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EXPERIMENTAL INVESTIGATION OF THICK TARGET BREMSSTRAHLUNG RADIATION PRODUCED BY ELECTRONS OF 1.00, 1.50 AND 2.00 MeV

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(*Received* 14 *December* 1959)

Abstract—Measurements have been made of the 0° bremsstrahlung spectrum emerging from thick targets of polystyrene, aluminum, copper, silver and gold bombarded by mono-energetic electrons of 1.0 MeV, 1.5 MeV and 2.0 MeV. The measurements were made using a 4 in. diameter by 4 in. long NaI(Tl) scintillation detector unit with a lead shield and collimator. The pulse-height spectra were recorded in a twenty-channel, pulse-height analyzer and were converted to photon spectra using an experimentally determined response matrix for the scintillation detector. Corrections and errors are discussed and the results are plotted as families of curves of number of photons per mega-electron volt per microampere per steradian at zero degrees vs. photon energy. Comparisons are made with some elementary semi-empirical predictions of thick target bremsstrahlung spectrum shapes.

INTRODUCTION

THE intensity and energy spectrum of bremsstrahlung radiation generated when electrons are stopped in thick targets is a function of (a) the electron velocity, (b) the atomic number, Z, of the target, and (c) the angle between the direction of observation and the incident electron beam.[1,2] Knowledge of the manner in which the intensity and spectrum depend on these variables is necessary to the complete understanding of the problem of bremsstrahlung production.

Experimental investigations of bremsstrahlung radiation produced by electrons in the energy range of a few mega-electron volts[3–10] have provided extensive information about the gross characteristics, i.e. half-value layers, roentgen output, etc., of the continuous X-ray beams from thick targets of materials of different atomic numbers. However, information to date about the spectrum of photons produced by electrons in this energy range[10–21] has not covered the entire region of interest. This problem has received theoretical treatment,[22–28] but it has been limited in scope.

This limited theoretical treatment of thick target bremsstrahlung spectra arises largely from the fact that the spectral distribution of the radiation leaving a thick target results from the superposition of several complex processes and, at present, information on each of these individual processes is far from complete. The most important of these processes are: (1) Radiation by mono-energetic electrons incident on a thin target; this is given by the differential bremsstrahlung cross-section (i.e. differential in photon energy, photon angle and electron energy). (2) Electron penetration, scattering and energy degradation in the target. (3) Absorption of bremsstrahlung photons in the target.

There have been some attempts to calculate bremsstrahlung spectra in an approximate manner by use of the following techniques: (1) Application of the Bethe–Heitler theory to calculate the bremsstrahlung production for the thin target case and then correction of the obtained theoretical distribution of photons for the absorption by the target itself[29] (the resulting spectrum is accurate to within about

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1

20 per cent).[30] (2) By assuming the thick target spectrum to be the summation of a series of thin target spectra of varying incident electron energies and numerically integrating a number of these thin target cases.[1] (3) Correction of the Bethe–Heitler thin target bremsstrahlung formula for the use of the Born approximation by applying a factor derived from SOMMERFELD's exact calculation. This corrected thin target spectra was integrated yield the true bremsstrahlung spectra emitted from the target.

EXPERIMENTAL METHOD

The NRDL Van de Graaff accelerator was used as the source of mono-energetic electrons, calibrated relative to $Be^9(\gamma, n)Be^8$ threshold taken at 1662 keV and the $Li^7(p, n)Be^7$ threshold taken at 1881.4 keV. It was not possible to use the $H^2(\gamma, n)H^1$ reaction with a threshold at 2.17 MeV to obtain a second electron calibration point with this Van de Graaff

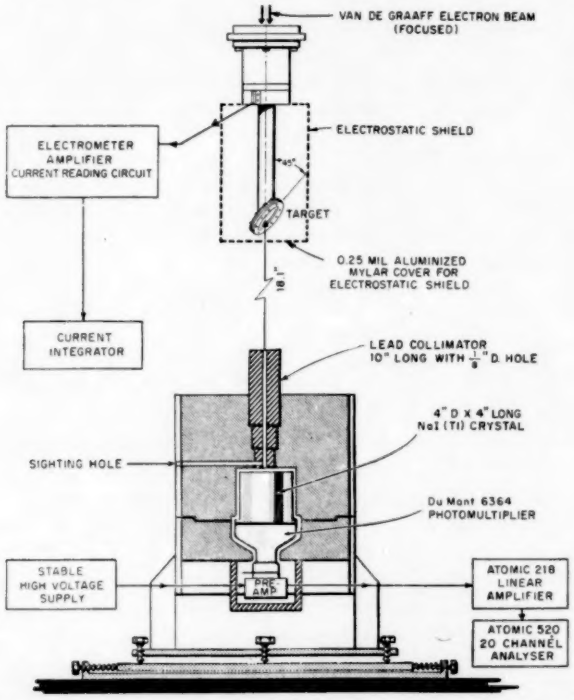
VAN DE GRAAFF ELECTRON BEAM (FOCUSED)
ELECTROSTATIC SHIELD
ELECTROMETER AMPLIFIER CURRENT READING CIRCUIT
45°
TARGET
0.25 MIL ALUMINIZED MYLAR COVER FOR ELECTROSTATIC SHIELD
CURRENT INTEGRATOR
LEAD COLLIMATOR 10" LONG WITH 1/8" D. HOLE
4" D X 4" LONG NaI (Tl) CRYSTAL
SIGHTING HOLE
Du Mont 6364 PHOTOMULTIPLIER
STABLE HIGH VOLTAGE SUPPLY
PRE AMP
ATOMIC 218 LINEAR AMPLIFIER
ATOMIC 520 20 CHANNEL ANALYSER

FIG. 1. Schematic diagram of the Van de Graaff target and scintillation spectrometer.

taking account of energy loss along the path of the electron.[31]

In the present work measurements are made of the bremsstrahlung spectra from low-, intermediate- and high-Z thick targets for electrons with initial kinetic energies of 1.00, 1.50 and 2.00 MeV. The emitted bremsstrahlung spectra were measured using a multichannel scintillation spectrometer. The experimentally obtained pulse-height distributions were then subjected to a series of corrections to

accelerator since the maximum energy of the machine is 2.0 MeV. However, it was felt that the generating voltmeter, which indicates accelerator voltage, could be calibrated reliably using positive ion acceleration and that the calibration obtained thereby was applicable to electron operation. The long-time energy stability of the generator was approximately 2 per cent. A schematic diagram of the experimental arrangement is shown in Fig. 1. Geometry considerations indicate that the angular spread in the beam is less than 0.3°. Measurement of the electron beam size at the target by visible darkening of thin

Table 1. Range and path length values for mono-energetic electrons in the target materials

Target material	Electron energy	Target thickness (in.)	Slant thickness (in.)	Density (g/cm³)	Slant thickness (g/cm²)	Path length		Extrap. range R_0† (g/cm²)
						Resid. range[32] (g/cm²)	Resid. range* (g/cm²)	
Styrene C_8H_8	1.0	0.104	0.147	1.06	0.396		0.439	
	1.5	0.168	0.237		0.638		0.708	
	2.0	0.231	0.326		0.878		0.976	
Aluminum $Z = 13$	1.0	0.056	0.079	2.7	0.542	0.543	0.545	0.412
	1.5	0.090	0.128		0.878	0.872	0.876	0.678
	2.0	0.123	0.174		1.193	1.198	1.200	0.945
Copper $Z = 29$	1.0	0.019	0.027	8.9	0.611	0.634	0.634	
	1.5	0.032	0.045		1.018	1.013	0.974	
	2.0	0.043	0.061		1.380	1.387	1.39	
Silver $Z = 47$	1.0	0.0190	0.0269	10.5	0.718	0.720	0.705	
	1.5	0.0304	0.0430		1.150	1.147		
	2.0	0.0415	0.0587		1.568	1.567		
Gold $Z = 79$	1.0	0.012	0.017	19.3	0.833	0.827		
	1.5	0.019	0.027		1.322	1.312		
	2.0	0.026	0.037		1.810	1.787		

* Ann T. Nelms, *NBS Circ.* 577 (1956).
† L. Katz and A. S. Penfold, *Rev. Mod. Phys.* **24,** 28 (1952).

glass lantern slides indicates a diameter of about 2 mm for all energies used. The Faraday cup assembly holding the target is separated from the end of the accelerator vacuum system by a lucite insulator. It was fabricated of thick-wall aluminum ($\frac{3}{4}$ in. inside diameter, $11\frac{1}{4}$ in. length) to prevent the escape of secondary electrons and enclosed in a grounded thin metal container for electrostatic shielding. The present measurements are concerned only with observations of the radiation from gold, silver, copper, aluminum and polystyrene. The target holder was designed to position the various targets at an angle of 45° to the incident electron beam to permit observation of the emitted bremsstrahlung at various angles from 0° to 120°. However, all the measurements were made for an angle of zero degrees between the collimator spectrometer axis and the direction of the incident electron beam. The target thickness in the direction of the incident electron beam, for each element, at each energy, was equal to the calculated maximum residual range of the electron at that energy, using Spencer's values of residual range.[32] Residual range as defined by Spencer is equivalent to maximum electron path length based upon an integration from zero to the energy of the incident electron of the $(dT/dr)^{-1}$ values for electrons in the medium. Table 1 gives the target thicknesses used, the target thickness in the incident electron direction, and the residual range values of Spencer[32]. While the actual target thickness is less than the residual range, it was felt that the small fractional solid angle into which the electrons must go at 45° to the incident beam direction in order to get all the way through the target would make this possible electron loss negligible. Neither the fraction of electrons scattered backwards out of the target proper, nor that which might have penetrated the whole target thickness was determined experimentally.

The bremsstrahlung from the target was detected and measured using a total absorption scintillation spectrometer.[33] The particular spectrometer used in this experiment has as a photon detector, a 4 in. diameter by 4 in. long NaI(Tl) crystal unit optically coupled to a 5 in. diameter Dumont 6364 photomultiplier tube. These components are mounted inside a steel-encased lead shield and radiation

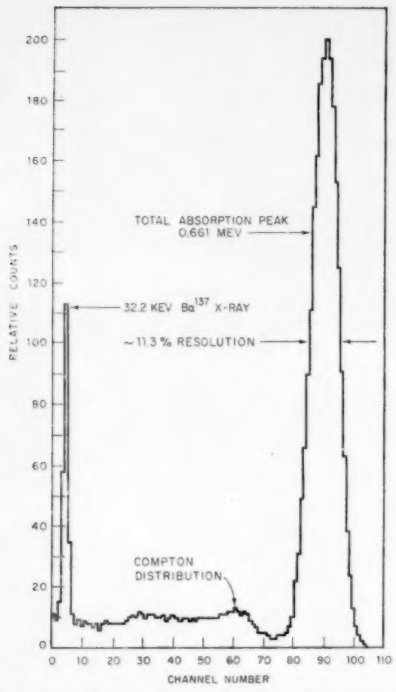

FIG. 2. Pulse-height spectrum from the 4 in. diameter by 4 in. long NaI(Tl) crystal spectrometer using the collimated γ-rays from a Cs^{137} source.

collimator system which is shown in Fig. 1. The detector shield and collimator system is fitted with an optical arrangement which enables one to sight in through the side of the shield, on to a 45° mirror which can be placed at the crystal end of the collimator, and then up the collimator axis toward the radiation source. This makes it possible to aim the unit directly toward the target. The lead collimator plug in front of the crystal is 10 in. long and has a $\frac{1}{8}$ in. diameter coaxial hole. This collimator can be removed and replaced with a solid plug for background measurements.

The amplified signals from the photomultiplier were analyzed in a twenty-channel, pulse-height analyzer. A precision sliding pulser was used to position the twenty channels in each of six different adjacent positions which enables one to cover a 60 V pulse-height spectrum with 120 half-volt channels. The pulser also enables one to adjust the twenty channels to equal width.

Eleven mono-energetic γ-ray sources emitting γ-photons from 0.145 MeV to 2.76 MeV were used to evaluate the pulse-height response of the detector. The sources were placed as nearly as possible to the electron target position. The geometrical positions of the sources relative to the collimator-scintillation spectrometer were similar enough to that of the electron target so that the detector response accurately indicated response to target bremsstrahlung. A typical response spectrum for the mono-energetic γ-ray from Cs^{137} is shown in Fig. 2.

The bremsstrahlung beam emerged from the target and traversed 28.1 in. of air, 0.010 in. of aluminum, and a negligible amount of MgO before entering the NaI crystal. The photon spectrum measured in the crystal was corrected for the attenuation in the aluminum. The total aluminum attenuation coefficient was used, although this somewhat over-corrects at the high-energy end owing to the poor geometry of the situation.

Owing to photon penetration through materials, the collimator aperture appears to have a different size opening for the γ-rays of different energy emanating from the target (considered to be a point source on the collimator axis). The so-called penetration aperture, P (cm^2), is the increase in effective aperture due to collimator penetration. A theoretical treatment of the penetration aperture has been given by MATHER[34] and the results of this theory as applied to this experiment are shown in Fig. 3. The effective aperture is the sum of P and the geometrical aperture A (cm^2), and this sum divided by A is plotted as a function of energy in Fig. 3. This number is therefore the normalizing divisor for the photon spectrum to correct for collimator penetration. The geometrical aperture subtends an angle of 15′ at the target.

The resolving time of the twenty-channel analyzer is 2.3 μsec for the entire spectrum of pulses accepted, and the individual channel resolving time is 20 μsec. At the counting rates used in the experimental runs, counting loss in the total spectrum and in individual channels amounted to less than 1 per cent in the worst case and no correction is made for this effect.

The electron-beam axis was established by the spot burned by the electron beam on the glass disk placed at the target position and the lighted Van de Graaff filament, as seen through the glass. Using the sighting mechanism on the detector shield, and the positioning mechanism, the detector was placed so that the collimator axis coincided with the electron-beam axis.

The energy scale for the spectrometer was determined by calibrating the unit before each run with a Zn^{65} source (1.11 MeV) placed near the NaI crystal collimator. This calibration permitted adjustment of the amplifier gain to compensate for drift and also provided an absolute energy scale for the

scintillation spectrometer since the location of the full-energy peak in the spectrometer was the same for the Zn^{65} source mentioned as it was under the geometrical conditions of the bremsstrahlung measurement. The gain stability of the system was checked by recalibrating with the source after each run, and in no case was it necessary to correct for gain shifts during the runs.

The gain of the analyzer system was adjusted so that forty channels of the analyzer would cover the energy range from about 20 keV up to the maximum energy of the particular bremsstrahlung spectrum being investigated (as determined by the electron-beam energy). Since only twenty channels of data could be taken at one time, each spectrum run consisted of four runs at one amplifier-gain setting and two runs at a high-gain setting. The runs were taken in the following manner: The twenty-channel analyzer was set to cover the pulse-height interval from 0–10 V, and an open collimator run was made with the electron beam on the target. The total charge collected on the target during the run was measured using the electrometer amplifier current monitor and integrator, and the time for a run was recorded. Beam current had been adjusted so that the total spectrum and individual channel counting rates fell below values which would require counting loss corrections. A background was then taken at the same analyzer setting and for the same amount of integrated beam current by replacing the detector collimator with a solid plug. The twenty-channel analyzer was then set to cover the range from 10–20 V and open collimator and closed collimator runs were made for the same integrated beam current. In order to examine the low-energy portion of the bremsstrahlung spectra, including the characteristic X-rays produced in the high-Z targets, some high-gain runs were taken. In these runs the overall gain of the system was increased so that the twenty channels covered the energy interval from 0–220 keV. For these runs, an uncollimated Hg^{203} source (0.279 MeV) was used to calibrate the energy scale of the spectrometer. A correction was made for the difference in the position of the full-energy peak in a scintillation spectrometer for a collimated source from that of an uncollimated source. This correction was determined experimentally for several mono-energetic γ-ray sources. Both open collimator and background runs were made for this gain and these were normalized to the low-gain runs by running for the same integrated beam current.

It was determined experimentally that bremsstrahlung produced at places in the system other than in the target and visible to the detector was negligible.

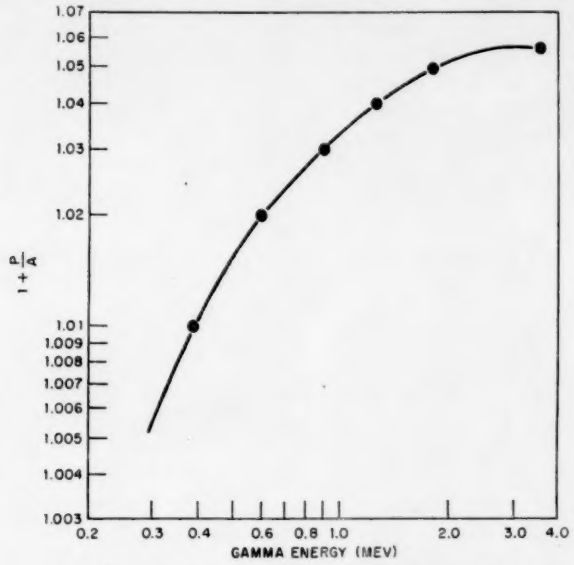

FIG. 3. Correction factor for penetration of the collimator by bremsstrahlung. $P(cm^2)$ is the increase in aperture due to penetration and $A(cm^2)$ is the geometrical aperture.

It was determined experimentally that bremsstrahlung produced at places in the system other than in the target and visible to the detector was negligible.

DATA REDUCTION

The data reduction consists of so manipulating the recorded pulse-height data that the complex photon spectrum originally incident upon the crystal (or a close approximation to it) may be derived.

There are four parts to this process: (1) the determination of the shape of the pulse-height spectra in the detector for mono-energetic sources at as many energies as are available in the region of interest; (2) the calculation of a grid of energies sufficiently fine to resolve any desired spectral structure; (3) the interpolation between these mono-energetic source spectra to obtain the corresponding shapes of the response function at each of these grid energies; and (4) the use of the matrix representing these interpolated spectra (either directly or by means of its inverse) to correct the spectrometer data for non-uniqueness and efficiency.

The data-reduction methods used in this paper follow closely those used by other investigators in the field.[12,17,33,35–37] The principal differences in the method to be described are: (1) a finer grid of energies is used; (2) more calibration sources are used; (3) details of a method of graphical interpolation between the source spectra are given; and (4) a comparison of unfolding by stripping and by inverse matrix multiplication has been made.

The observed spectrometer pulse-height distribution, $C(V)$ (number of voltage pulses per analyzer channel), that results from the spectral distribution of photons, $N(E)$ (number of photons per MeV), incident upon the crystal can be expressed as:

$$C(V) = \int_0^{E_{max}} R(V, E) N(E)\, dE \qquad (1)$$

where:

V is the pulse height;
E is the energy of the photon;
$R(V, E)$ is the response function of the detector;
$R(V, E)\, dE$ is the probability that a photon of energy E will produce a pulse of height between V and $V + dV$, i.e. $N(E) = 1$.

The function $R(V, E)$ is experimentally determined by recording the pulse-height distribution, $C(V)$, for a number of mono-energetic γ-ray-emitting nuclides. Hence:

$$C_k(V) = N_k R(V, E_k)$$

$$\text{where} \quad N_k = \int_0^E N(E)\, \delta(E_k - E)\, dE \qquad (2)$$

and $\delta(E_k - E)$ is the Dirac delta function.

Response spectra obtained as indicated earlier form the basis for the response matrix. After subtracting background, any extraneous peaks in the response pulse-height spectra were subtracted (e.g. annihilation radiation in Na^{22} and Zn^{65}, X-ray peaks in Hg^{203}, Cs^{137}, etc.). The lower tail of the total absorption peak was constructed.

For interpolation purposes the photopeaks were approximated by Gaussian distributions having the same parameters (e.g. mean pulse height and standard deviation σ). For a comparison with other investigations it is more convenient to use resolution (full width at half maximum divided by the mean). The relationship between resolution, R, and σ for a Gaussian curve is given by $R = 2.355\ \sigma/V'$ where σ and V' (mean pulse height of total absorption peak) are expressed in the same units. The curve of resolution vs. source energy is shown in Fig. 4. Fig. 5 gives the curve of detector efficiency calculated using $\varepsilon = 1 - \exp[-\mu(E)x]$ where $\mu(E)$ is the linear absorption coefficient[38] for photons of energy, E, in NaI and x is the crystal length, 4 in. The photofraction, p, for each response curve was calculated by dividing the area under the total absorption peak by the total area. This photofraction is shown plotted in Fig. 6 as a function of source energy.

Although the percentage resolution of the spectrometer becomes poorer for low-energy photons the absolute resolution becomes better. This fact together with the peaking of the bremsstrahlung spectra at low energies suggested the use of uniformly varying channel widths for the response matrix—narrow at the low-energy end and broad at the upper end.

Upon plotting the σ's as a function of source energy it was found that a straight line could be drawn through most of the experimental points. This indicates a linear dependence of σ with energy for this crystal. Normally it has been

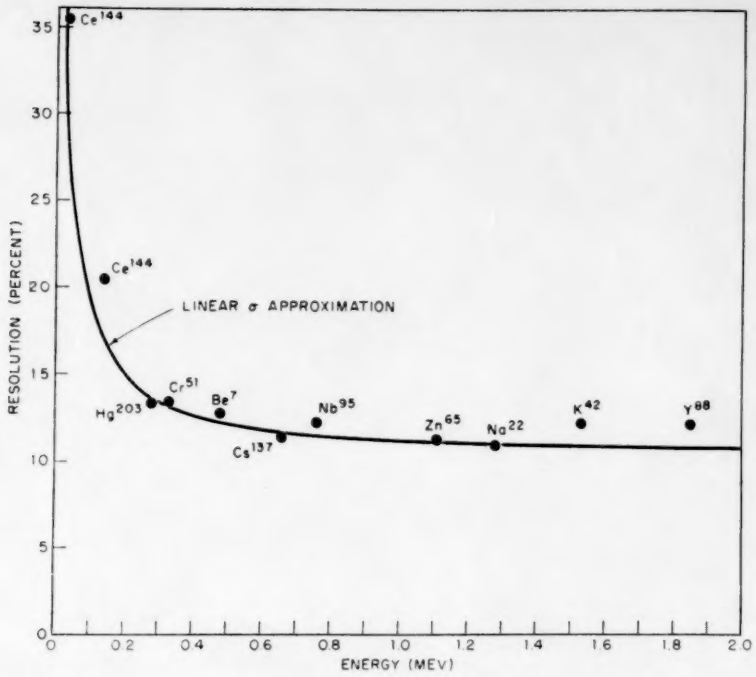

FIG. 4. Spectrometer resolution, $R = 2.355\ \sigma/V'$, vs. source energy. σ is the standard deviation of the total absorption peaks in pulse-height units and V' is the mean pulse height associated with the total absorption peaks. The linear approximation is evident in a plot of σ vs. V'.

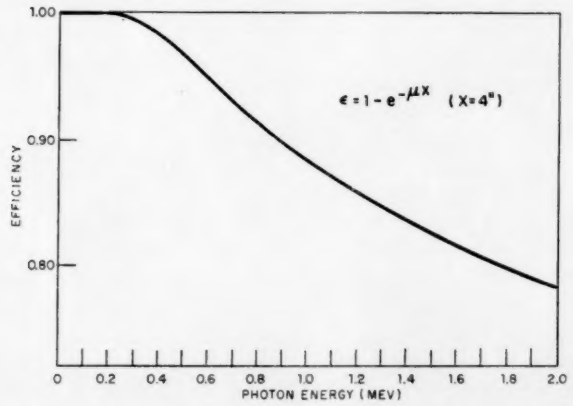

FIG. 5. NaI(Tl) crystal detection efficiency vs. photon energy. μ is the total attenuation coefficient of NaI(Tl) and ε represents the efficiency for all pulses in the pulse-height spectrum due to mono-energetic γ-rays.

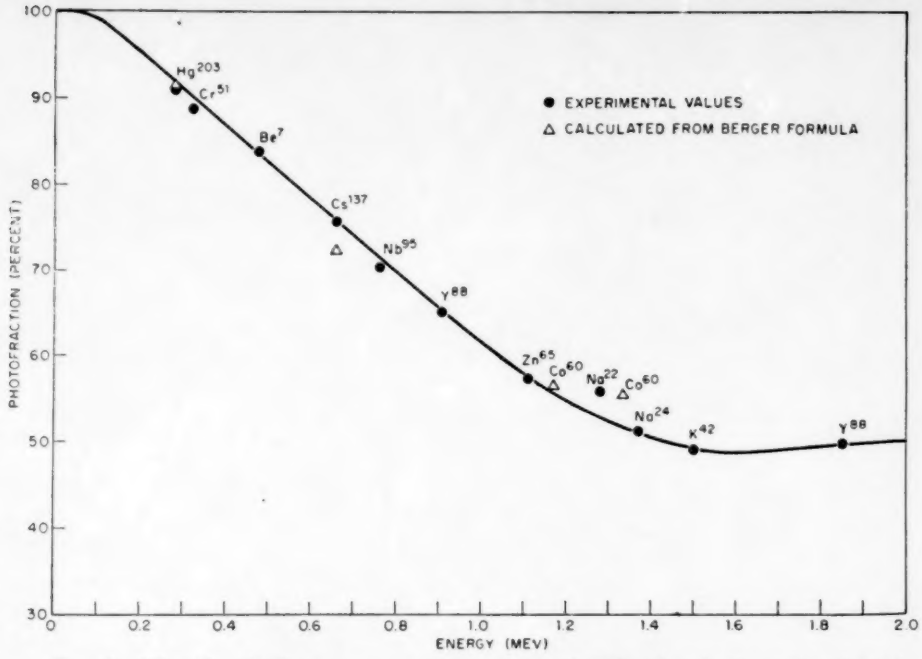

FIG. 6. "Photofraction" vs. source energy where photofraction is the ratio of total absorption peak counts to all the counts in the pulse-height spectrum.

found that σ^2 is proportional to photon energy (E). The rather anomalous behavior of this crystal is apparently due to a light production and collection efficiency which is different from the ideal case. Each channel width was made proportional to the σ computed by means of the linear approximation formula using the energy assigned to the center of that channel. The value of this proportionality constant was determined by the maximum energy to be determined (2.0 MeV), the condition that adjacent channels should have one edge in common and, finally, that the number of channels should not be much more than 50.

This resulted in a 55-channel grid covering an energy range of from 0–2.0 MeV with channel widths increasing from 9 keV–96 keV.

After subtracting the total absorption peaks, the Compton distribution area was normalized to $\varepsilon(1 - p)$ and replotted. The ordinates of the points of intersection of each of these curves with the matrix pulse-height grid lines were next tabulated. From this table a family of interpolation curves parametric in matrix pulse-height channel number was plotted. Each curve shows the variation of ordinate values (normalized counts per channel) with source energy for a given pulse height. In a similar manner the interpolation curves were sliced at the matrix energy grid values. The ordinate values at these intersection points were tabulated, completing the Compton interpolation process. The area of the interpolated Compton distribution histograms for each "grid source" energy was now renormalized, where necessary, to $\varepsilon(1 - p)$.

The final step in forming the response matrix was the regeneration of the total absorption peaks corresponding to each of these fifty-five Compton distributions. For each matrix channel a Gaussian histogram centered at that channel and having the standard deviation associated with that channel was calculated. The area normalization was to the $p\varepsilon$ value for that channel. When these normalized histogram ordinates were added to the appropriate Compton histogram values the response matrix was completed. Summing over pulse height for any

one of the matrix columns thus yielded the value of ε for the energy associated with that column.

Each element, r_{ij}, of the square response matrix gives the probability of recording a pulse of height between V_i and $V_i + \Delta V_i$ for a γ-ray of energy between E_j and $E_j + \Delta E_j$ incident upon the center of the crystal face.

The experimental bremsstrahlung pulse-height distributions were plotted on semi-log paper and a smooth curve drawn through the points. Next, the values of the curve at its points of intersection with the matrix pulse-height channel edges were read. From these, the value of a histogram ordinate giving the same area as that in the matrix channel under the curve was calculated. To reduce these pulse-height spectra to γ-number spectra, two equivalent processes may be utilized. The inverse matrix may be calculated and multiplied by the pulse-height distribution to yield the incident spectra:

$$[R^{-1}]_{ji}C_i = N_j$$

or the response matrix R_{ij} may be used to determine N_j by an iterative process. The one-sidedness of the response function permits a stripping of response curves from the upper edge by a method similar to that of STEPHENSON and BELL[39]. It consists of determining a series of normalizing factors (N_j) which, when multiplied by the synthesized response histograms R_{ij}, will approximate the pulse-height distribution C_i. Fig. 7 shows an example of this type of unfolding. The dashed curve is the experimental pulse-height histogram (bremsstrahlung spectrum 0°, 1 MeV electrons on thick gold target). The solid curve is the histogram yielded by inverted matrix multiplication while the shaded curve is the histogram calculated by the stripping method. It can be seen that, in general, the agreement between the

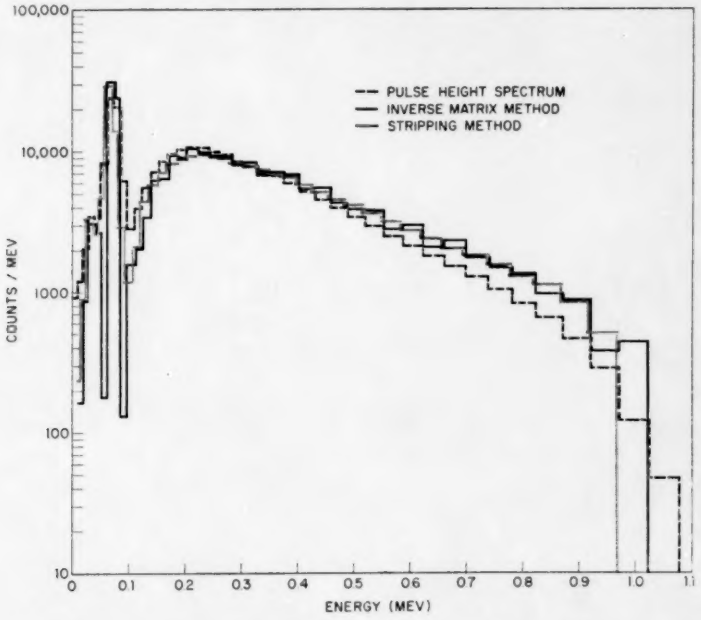

FIG. 7. Example of results of the stripping method for spectrum unfolding compared to inverse matrix multiplication. Comparison is made to the original pulse-height spectrum.

two methods is good. Inverse matrix multiplication was used for all the data described in the next section.

RESULTS OF MEASUREMENTS

The results of the experiment are shown in Figs. 8, 9 and 10 as families of differential

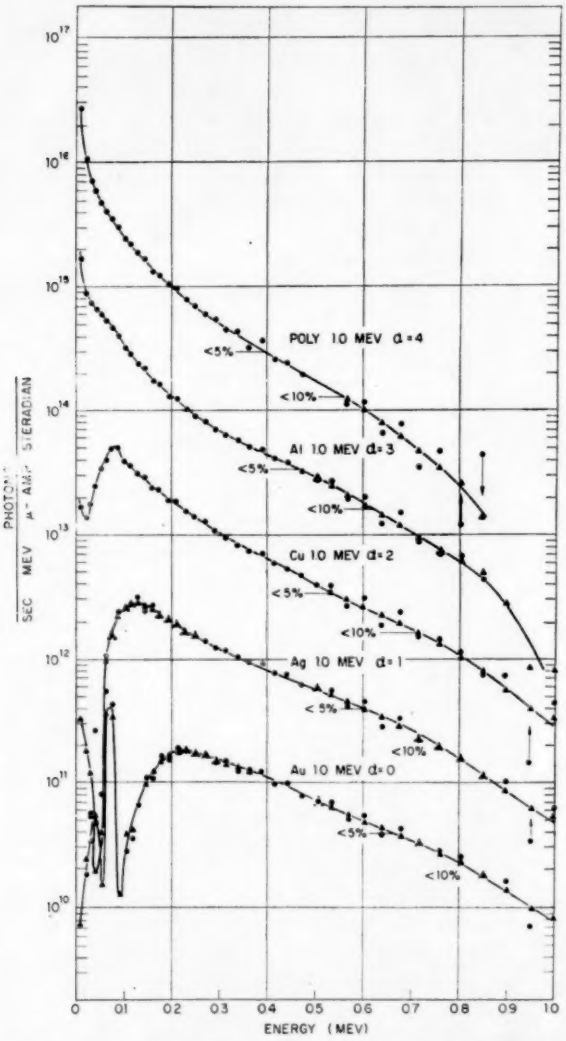

FIG. 8. Family of bremsstrahlung spectra for 1.0 MeV electrons on thick targets of different Z. The measurement is made at zero degrees relative to the incident electron direction. Each curve is multiplied by 10^{α} for clarity of presentation. The triangles represent points obtained from the data points (dots) by use of a graduating formula. Arrows indicate to which curve a point belongs. Percentage values represent points on curves above which the statistical and binning errors are less than the indicated values.

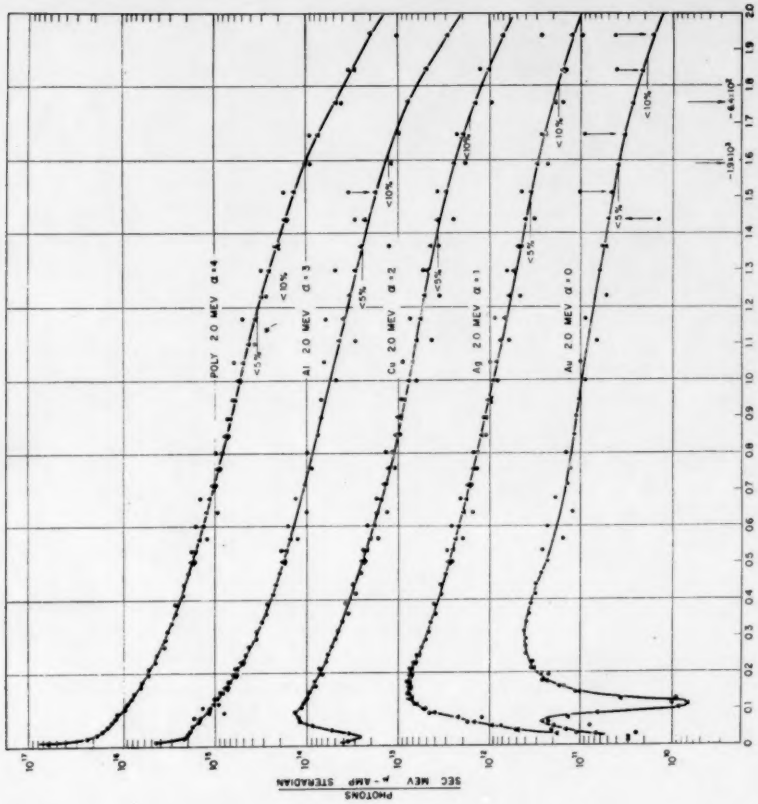

FIG. 10. Family of bremsstrahlung spectra for 2.0 MeV electrons on thick targets of different Z. The measurement is made at zero degrees relative to the incident electron direction. Each curve is multiplied by 10^{2} for clarity of presentation. The triangles represent points obtained from the data points (dots) by use of a graduating formula. Arrows indicate to which curve a point belongs. Percentage values represent points on curves above which the statistical and binning errors are less than the indicated values.

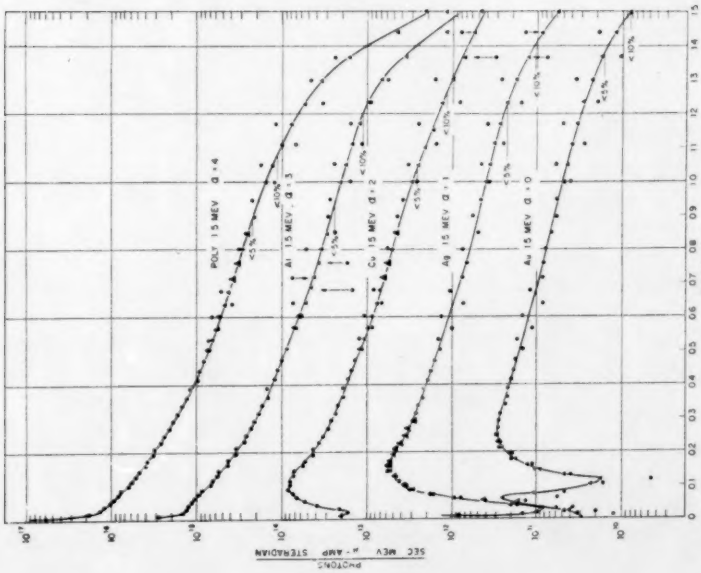

FIG. 9. Family of bremsstrahlung spectra for 1.5 MeV electrons on thick targets of different Z. The measurement is made at zero degrees relative to the incident electron direction. Each curve is multiplied by 10^{2} for clarity of presentation. The triangles represent points obtained from the data points (dots) by use of a graduating formula. Arrows indicate to which curve a point belongs. Percentage values represent points on curves above which the statistical and binning errors are less than the indicated values.

energy spectra for the bremsstrahlung emerging at zero degrees relative to the incident electron beam from targets of different Z bombarded by mono-energetic electrons at different energies. The unfolded spectra were corrected for the absorption of γ-radiation in the aluminum top of the crystal container and for the collimator penetration by γ-rays. The left-hand ordinate of Figs. 8–10 is in units of register counts (scale of 16) per MeV per 1000 $\mu\mu$A min per 1.5×10^{-5} sterad at 0°. To convert to the right-hand scale, which is in units of photons per sec per MeV per μA per sterad at 0°, the ordinate values must be multiplied by $(1.778 \pm 0.018) \times 10^7$. The fluctuations of some of the points of a given spectrum about the smooth curve are introduced by the multiplication by the inverted response matrix. Points on the smooth curve (indicated by triangles) in the region of oscillation were obtained by use of a graduating formula:

$$y_i = \tfrac{1}{16}(y_{i-2} + 4y_{i-1} + 6y_i + 4y_{i+1} + Y_{i+2})$$

This is approximately equivalent to folding the data by a Gaussian. To see any possible smearing caused by this technique it was applied to the X-ray region of the 2 MeV Au spectrum. Comparing the smoothed curve of the 2 MeV Au data and the non-smoothed curve of the 1.5 MeV Au data in the X-ray region there seems to be little smearing.

SUMMARY OF ERRORS

(1) The correction for absorption in the aluminum top of the crystal container only affected the first few channels. This correction is known to better than 2 per cent and at most corrects the data by 18 per cent. The correction error is therefore 0.4 per cent at most.

(2) The maximum correction for collimator penetration amounted to 5 per cent at channel 55. This correction is known to better than 10 per cent so that the correction error is at most 0.5 per cent.

(3) All spectra run at other values of integrated beam current were corrected to (1000 ± 5) $\mu\mu$A min. The absolute calibration is (1000 ± 10) $\mu\mu$A min.

(4) Geometrical errors.

(a) The position, size and incident angle of the electron beam determine the effective angle of detection and the solid angle subtended by the detector. The target size and position were determined to within 0.3 mm. With this size and position limit, there was no significant error introduced in the absolute results. The incident angle of the electron beam is determined to within 0.05° so that there is a negligible error in the results due to this angle error.

(b) The target thickness was determined to ± 0.001 in. The purity of the targets used has been determined by spectrochemical analysis and the impurities were small enough that no correction for impurities was required.

(c) The detector axis alignment error is estimated to be 0.4° while the collimator acceptance half angle is approximately 0.2°. Due to the approximately isotropic angular distribution of thick target bremsstrahlung, no significant error is introduced in the absolute value of the results.

(d) The geometrical solid angle subtended by the collimated crystal was $(1.500 \pm 0.007) \times 10^{-5}$ sterad.

(5) Counting and data manipulation errors.

(a) The statistical errors of the experimental measurements are projected through the binning and unfolding processes and appear magnified to a greater or lesser degree in the unfolded bremsstrahlung spectra. Estimated 5 per cent and 10 per cent levels of probable error have been indicated for each spectrum. These include the statistical and binning errors but *not* those due to unfolding.

(b) The analyzer channel widths were periodically checked throughout the course of the experiment and found to be equal to within ± 1 per cent.

(c) An analysis of the total counting rate levels in the analyzer yielded a dead time loss of 0.4 per cent for the highest count rate measured.

(6) Energy calibration. The gain stability checks indicated a gain stability of 0.2 per cent between calibrations. The accuracy of the energy assignment to pulse height is $\sim$1.0 per cent.

DISCUSSION

As might be expected from thin target theory, the shapes of all the bremsstrahlung

spectra in Figs. 8–10 are quite similar. (For clarity, each of the curves has been displaced (multiplied) in ordinate by a factor of 10^x.) Those differences in shape which are present are due primarily to the difference in electron angular and energy distributions as a function of fractional range in the different targets and the variation in absorption of photons by the target themselves. Thus in the case of the low-Z materials, such as polystyrene and aluminum, where only the high-energy electrons contribute to bremsstrahlung production and photon absorption is a minimum, the shapes of the 1.00, 1.50 and 2.00 MeV curves are similar to those observed in the case of thin targets. In the case of the higher-Z materials, such as copper and silver, the increase in target self-absorption and the increase in bremsstrahlung production becomes evident. For these two elements the 1.00, 1.50 and 2.00 MeV curves have the typical thick target knee-shaped appearance. In the case of the 1.00 MeV silver spectrum the characteristic silver X-ray lines (22–27 keV) are displayed as a single peak. The 1.00, 1.50 and 2.00 MeV gold spectra show marked thick target shapes with the low-energy characteristic gold X-ray lines (70–79 keV) shown as one peak. The second small peak in the 1.00 MeV gold spectrum, some 30 keV below the characteristic X-ray peak, is the iodine escape peak. This peak, which is due to the NaI detector, was not removed by the inversion matrix used to transform the experimental pulse-height distribution to the true spectrum since the matrix did not contain sufficient detail at this very low energy to remove that portion of the detector response. It is therefore not a legitimate part of the source spectrum.

Table 2. Energy values at the bremmsstrahlung spectrum peaks

	1.00 MeV	1.50 MeV	2.00 MeV
	Peak energy (keV)	Peak energy (keV)	Peak energy (keV)
Cu ($Z = 29$)	80	105	85
Ag ($Z = 47$)	130	165	165
Au ($Z = 79$)	230	280	295

Table 2 shows the peak energies for those spectra which show the typical thick target shape. It is interesting to note that in the case of gold, the peak is shifted only some 65 keV from 230 keV at 1.00 MeV to 295 keV at 2.00

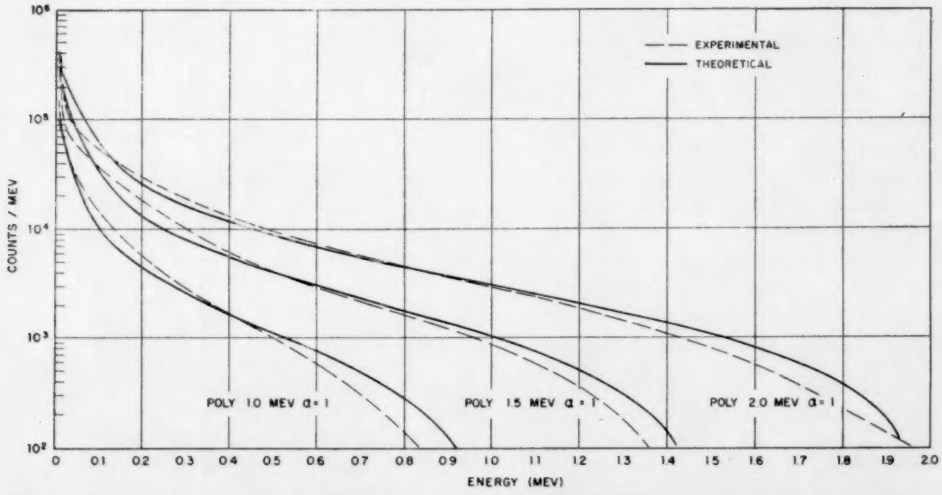

FIG. 11. Comparison of some of the experimental results (dashed line) with the results obtained using the semi-empirical formula of KULENKAMPFF (solid line—indicated as theoretical).

MeV. This is considerably less than the 1.1–1.3 MeV peak often shown in the supervoltage medical X-ray literature. Experimental results not reported in this paper show that the addition of 0.125 in. thickness of gold to the 2.00 MeV thick Au target shifts the peak at most to 530 keV. The anomaly in the case of the copper peaks is due to errors in drawing a smooth curve through the experimental points.

KULENKAMPFF[40] has found that a simple relationship of the form $I = c(\nu g - \nu)$ can be used to predict an approximate shape for thick target bremsstrahlung intensity spectra in certain circumstances. In this expression νg is the end-point freqeuncy, ν the frequency at any point and c is an experimentally determined constant. This expression was used[40] to predict a shape which agreed well with experimental results, for the reflection spectrum coming out of a thick tungsten target at 90° for incident electrons from 20 to 50 keV. Fig. 11 shows a comparison of the experimental results in this experiment for the case of 1.0 MeV, 1.5 MeV and 2.0 MeV electrons on thick polystyrene targets with the spectra predicted by a formula of the type mentioned above. Since these are the 0° experimental spectra it would be expected that low-Z results might agree better with the theory for 0° transmission spectrum since there is little γ-ray self-absorption in the low-Z targets.

A more complete analysis of the experimental results and detailed comparison of experiment and theory must await a more comprehensive theoretical treatment than presently exists. It is hoped that this experimental study may serve to emphasize the need for a detailed theoretical attack on the problem of thick target bremsstrahlung spectra.

For a more detailed description of the experiment and analysis than is presented here, readers are referred to a laboratory technical report on this experiment.[41]

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THE UPTAKE AND ELIMINATION OF CESIUM-137 BY A GRASSHOPPER—*ROMALEA MICROPTERA**

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Abstract—Adults of *Romalea microptera*, the eastern lubber grasshopper, were fed cesium-137 in bean plants to investigate uptake and elimination of this isotope. A biological half-life of 4–5 days was obtained. In experiments where grasshoppers were allowed to feed repeatedly on Cs-contaminated food, the biological half-life was used to predict the equilibrium values. Most of the ingested Cs^{137} was concentrated in muscular tissue, and also occurred in the digestive tract and reproductive organs. Only trace amounts were found in the exoskeleton.

INTRODUCTION

THE bio-accumulation of fission products has received attention from health physicists, radioecologists and others who are interested in the biological consequences of environmental contamination resulting from radioactive waste disposal and fallout.[1,2] So far, much research effort has been expended on aquatic biological systems. Terrestrial biological systems have received attention directed towards evaluation of the hazard to humans, with emphasis placed on accumulation from the soil by crop plants and by domestic animals.[3]

Terrestrial invertebrates, although not in the direct food chain leading to man, play an important role in the biological cycling of minerals. Little is known of the rates of accumulation and elimination of fission products by these animals. The present paper presents the results of laboratory experiments on accumulation of cesium-137 by a species of large grasshopper. The studies were conducted to aid in the interpretation of field data collected in areas contaminated by waste disposal operations, where Cs^{137} and other fission products have become incorporated into biological systems.

MATERIALS AND METHODS

Adult *Romalea microptera* Beauv., the eastern lubber grasshopper, were purchased from Carolina Biological Supply Company, Elon College, North Carolina. During the experiments grasshoppers were kept in cages in controlled environment rooms at 70°F and about 70–80 per cent relative humidity.

Bean plants were used as the primary food source for the grasshoppers. Introduction of Cs^{137} into the grasshoppers' food was accomplished by injecting 1 μc Cs^{137} (0.2 ml aqueous solution) directly into the stems of the bean plants. Autoradiography showed that the cesium moved readily throughout all parts of the plant. For some of the uncontaminated feedings, native plants (*Polygonum* spp.) were substituted for the bean plants, since the grasshoppers required more plant food than could be grown readily.

A model 810 Baird–Atomic γ-scintillation detector with a 2 × 2 in. well crystal was used for measurement of Cs^{137} in plant materials, grasshopper tissues and living grasshoppers. Though living grasshoppers were too large to fit into the well, satisfactory counting (at 5 per cent efficiency) was achieved by placing the grasshoppers in cellophane sleeves, which held them in position on top of the crystal. The efficiency for this method was established by injecting known amounts of Cs^{137} into the abdominal cavities of test insects; experimental insects were counted subsequently at the same geometry.

* This research was conducted in the Health Physics Division, Oak Ridge National Laboratory, operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

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RESULTS

Rate of elimination

Elimination rate may be characterized in terms of a biological half-life, which is the time required for an organism to eliminate one-half of an administered dose. A biological half-life of about 4–5 days was estimated for Cs^{137} in *Romalea microptera*. This estimate was obtained by isolating several individuals which had reached high body burdens and by measuring their radioactivity at about 12 hr intervals. The results are shown in Fig. 1, where the average body burdens of five individuals are plotted as a function of time. While the 4–5 day rate is theoretically an effective half-life, the physical half-life of Cs^{137} is so long (about 33 years) that physical decay is of no significance in these measurements.

The estimate of a 4–5 day biological half-life is adequate for the central portion of the curve shown in Fig. 1. However, components of Cs^{137} eliminated with other biological half-lives may be present. Initially, elimination proceeds at a rate corresponding to a 2-day half-life. The last points on the curve suggest that a longer half-life component may be present also. Since virtually nothing is known of the metabolism of cesium by these insects, it is not yet possible to evaluate these components in terms of physiological phenomena. The use of a 4-day biological half-life proved adequate for the calculations presented here.

Equilibration with food

The accumulation of cesium by the grasshoppers was followed in forty-eight grasshoppers (three cages of sixteen grasshoppers each). Individual grasshoppers were identified by numbers painted on their backs. Bean plants containing a total of 2 μc Cs^{137} were placed in cage I on every second day, so that the average dose per grasshopper was 0.125 μc. Those in cage II received this dose every fourth day, and those in cage III every eighth day. In cages II and III cesium-free plants complemented the feeding so that each cage of grasshoppers was fed on every second day. All of the food placed in the cages was consumed.

When the equilibration experiments were started, the grasshoppers already contained

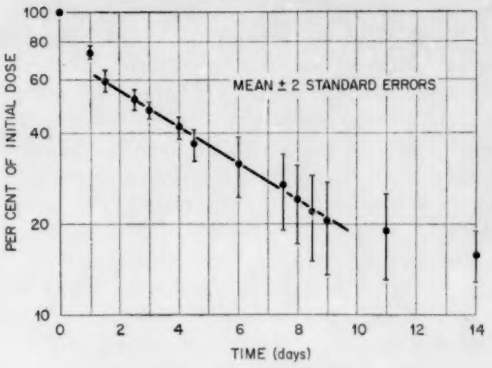

FIG. 1. Elimination of cesium-137 by *Romalea*.

some Cs^{137} from preliminary trials. Those grasshoppers with the least body activity were placed in cage I and received the most cesium, while those with the most body activity were placed in cage III and received the least cesium in food. Grasshoppers in cage II had an intermediate level of body activity and were placed on the intermediate diet. In this manner equilibration with food was followed both as an increase (cage I) and a decrease (cage III) in whole-body activity of the grasshoppers.

Results for cage III are illustrated in Fig. 2, which shows the mean whole-body activity of the grasshoppers as a function of time. The expected values were calculated from the equation

$$Q_t = Q_0 e^{-\lambda t}$$

where Q_t is the whole-body activity at time t, Q_0 is the initial whole-body activity, λ is $0.693/T_b$,

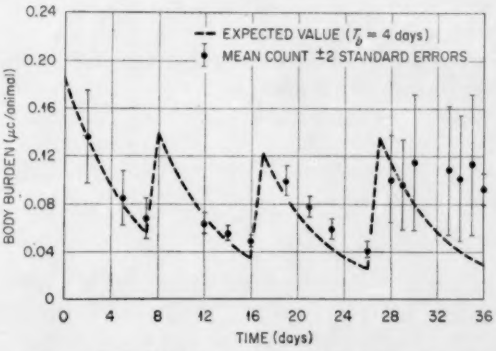

FIG. 2. Total body activity in *Romalea*. Cage III: 2 μc Cs^{137} every eighth day.

and T_b is the biological half-life. A T_b of 4 days was used in these calculations. The observed values show reasonable agreement with the expected ones for about 4 weeks, after which time the grasshoppers began to die. Seemingly, death was not the result of the experimental treatment since the control grasshoppers also began to die at about this time. The grasshoppers became lethargic several days before death, and did not feed or eliminate to any extent; thus, the observed radioactivity in them tended to remain constant. Accordingly, considerable deviation between expected and observed values at the end of the experiment could be anticipated. The standard errors shown in Fig. 2 became larger at the end of the experiment because the number of grasshoppers in the cage was reduced by deaths to three.

Grasshoppers in cage II showed little change in whole-body concentrations of Cs^{137} with time. The concentrations in the grasshoppers were almost at equilibrium with the food dosage when the equilibration experiments were started, so that little change in concentration could be expected.

Fig. 3 shows the results from cage I, where grasshoppers received 2 μc in bean plants every second day. These grasshoppers had the lowest initial level of body activity. The curve for expected values in Fig. 3 shows distorted periodicity due to an occasional missed feeding. Also the grasshoppers in this cage began to die after the first week and difficulty was experienced in attempts to adjust the dose so that the amount per grasshopper would remain constant. Agreement of observed with expected results is not good, but is reasonable for the first 2 weeks. In the latter part of the experiment the whole-body activities of the grasshoppers tended to remain constant, as did those in cage III. Presumably this stability of whole body concentrations was due to the cessation of feeding and elimination prior to death.

Concentrations in tissues

Analyses of grasshopper tissues and the use of autoradiograms showed that most of the ingested cesium is concentrated in the soft tissues. These findings are similar to those reported for vertebrate animals.[6] Table 1 presents the

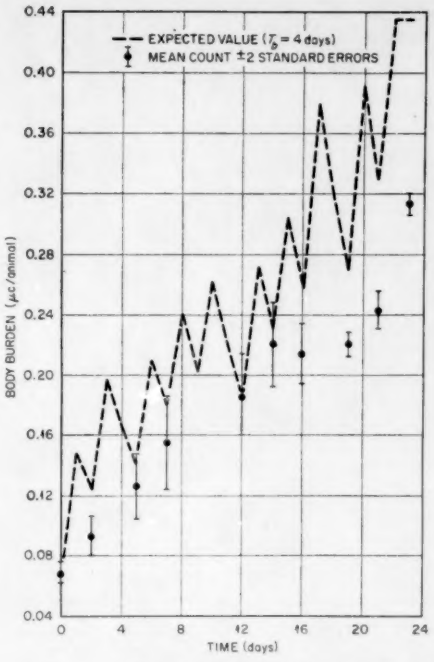

FIG. 3. Total body activity in *Romalea*. Cage I: 2 μc Cs^{137} every second day.

results of tissue analyses for four specimens of *Romalea microptera*. Muscular tissue, the digestive tract, and reproductive organs contained much more cesium than did the exoskeleton. "Residue" in Table I includes primarily a

Table 1. *Cesium*-137 *activities in body tissues of four* Romalea microptera
(Values are $\mu c \times 10^{-6}$ per mg)

Organ	Male cage II	Male cage II	Female cage II	Female cage III
Testes	258	320	—	—
Ovary	—	—	—	24
Eggs	—	—	135	10
Muscle	72	184	311	29
Digestive tract	178	248	705	26
Exoskeleton	3	5	2	3
Residue	66	87	138	15

Means of sample weights for these tissues were as follows: testes, 20 mg; ovary, 35 mg; eggs, 600 mg; muscle, 65 mg; digestive tract, 450 mg; exoskeleton, 100 mg; residue, 700 mg.

mixture of exoskeleton and muscle, which could not be separated.

DISCUSSION

The biological half-life of 4–5 days for Cs^{137} in *Romalea microptera* is similar to half-lives for other elements in other insects. GROSCH and SULLIVAN[4], working with phosphorus-32 in *Habrobracon*, found a biological half-life of 6–7 days (the half-life of 4–5 days reported in their paper is actually the effective half-life, since it includes the decay of P^{32}). GROSCH and LACHANCE[5] reported a biological half-life of less than 1 day for strontium-89 in *Habrobracon* females. Biological half-lives for P^{32} in *Drosophila* species of 2–3 days were reported by KING[7].

At least two other modes of elimination occur in addition to excretion and defecation. GROSCH and SULLIVAN[4] working with radiophosphorus in *Habrobracon* females found that oviposition is a major channel of elimination, since some 60 per cent of the radioactivity in these wasps was lost in this process. Females of *Romalea microptera* also would lose considerable radio-cesium by oviposition. The fraction would be considerably smaller than that reported by GROSCH and SULLIVAN, since most of the ingested cesium reached the muscular tissue and not the eggs. Another method of elimination in insects is the shedding of the exoskeleton during molting. In view of the low concentrations of cesium in the exoskeleton it seems unlikely that any significant amount of cesium would be lost during molting.

Cesium-137 concentrations in a grasshopper species such as *Romalea microptera* feeding on vegetation in a contaminated area evidently would come into equilibrium rapidly with the concentration in the vegetation. The time required to reach equilibrium was not measured directly in our experiments; however, time required to reach 50 per cent of the equilibrium value is the same as the time required to eliminate 50 per cent of the body activity for the theoretical case of constant intake and constant percentage elimination. Thus, the half-time for 50 per cent accumulation would be about four days, and equilibrium would be approximated (31/32 of the final value) after five such half-times.

In natural situations equilibrium may occur even more rapidly. The experimental animals were kept at a temperature of 70°F, which is probably lower than normal for *Romalea*. The food supply provided was probably a minimal diet for these insects. Also, all of the animals used were adults. The effects of higher temperatures should produce a higher rate of daily intake, as well as more rapid elimination of cesium. The grasshoppers might be expected to reach a higher body concentration more rapidly. And the immature stages, being smaller, might have a more rapid elimination rate than the adults.

The fraction of ingested Cs^{137} absorbed by the gut is necessary information for calculations of the turnover of cesium by insects in natural situations. This fraction may be found by comparing concentrations in food to concentrations in feces. In *Romalea*, however, feces were contaminated with excretion products. Feces from cage III had a higher per gram concentration of Cs^{137} than did the food, since the high initial Cs^{137} content of grasshoppers in this cage was eliminated with the feces. In mammals, ingested cesium is almost 100 per cent absorbed.[6] It seems unlikely that such high absorption occurs in *Romalea*, since fecal materials contained unbroken green cells of plant tissue, indicating incomplete digestion of the food.

The experiments on equilibration with food were not designed to detect radiation effects since none were anticipated at the low dosages involved. A possible explanation for the early deaths of the grasshoppers in cage I is the result of internal irradiation. The dosage delivered to the digestive tract was the highest delivered to any of the tissues, and this dosage did not exceed 50 rads. Little is known of the effects of internal irradiation in insects, but this dosage level is far below that required to produce effects from external irradiation. It is possible that the crop, or some other part of the digestive tract, received a higher dosage than did the tract as a whole. It is also possible that the alternate feedings of uncontaminated food served to flush out the digestive tracts of the grasshoppers in cages II and III, while the cage I grasshoppers received no such treatment.

Acknowledgements—We are grateful to Dr. S. I. AUERBACH and Mr. L. C. EMERSON, Health Physics Division, and Dr. M. A. KASTENBAUM, Mathematics Panel, Oak Ridge National Laboratory, for advice during the course of these studies.

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NEUTRON RESPONSE OF SILVER-ACTIVATED PHOSPHATE GLASS

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Abstract—The fluorescent response of silver-activated phosphate glass rods exposed to thermal and 14 MeV neutrons has been investigated. For thermal neutrons most of the absorbed dose in this glass arises from β-emission of activated silver nuclei, and the response per rad in this glass is the same as for Co^{60} γ-rays. For 14 MeV neutrons the response per rad is but 20 per cent of that for Co^{60} γ-rays. The possibility of using such dosimeters for simultaneous measurements of neutron and γ-ray intensities is discussed briefly.

In 1951, Schulman *et al.*[1] discovered that silver-activated phosphate glass irradiated by X-rays acquires the ability to fluoresce when excited by ultraviolet rays and that the amount of fluorescence is linearly proportional to the exposure dose at any particular quantum energy. Since then many studies of its application to radiation dosimetry have been made.[2–7] This paper is an extension to neutron response of the glass at two extremes of neutron energy—thermal neutrons and 14-MeV neutrons.

MATERIALS AND METHODS

The method used to measure induced fluorescence was essentially that of Schulman and Etzel[2]. Silver–glass rods 1 mm in diameter and 6 mm long were purchased from Bausch & Lomb Optical Co. Overall reading errors for the rods, including both the fluorometer error and the fluctuation in reading from rod to rod, were less than ± 5 per cent.

Thermal neutron exposure

The rods were exposed to the thermal neutrons in the thermal column of Brookhaven National Laboratory.[8] The neutron flux was measured by gold foils encased in an aluminum holder. Gamma contamination was ~100–115 reps/hr.[8] Two groups of rods were exposed: one in onionskin envelopes and the other encased in glass holders of 3-mm wall thickness.

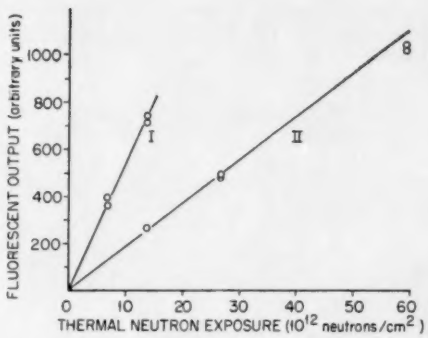

Fig. 1. Fluorescent output of silver-activated phosphate glass versus thermal neutron exposure. Fluorometer sensitivity: 136 units equal to the fluorescent output produced by a glass rod exposed to 1000 r of Co^{60} γ-rays at an exposure range of 100–350 r/hr. Neutron exposure rate range: 7.5–6.3 $\times$ 10^8 neutrons/cm^2 sec. Curve 1: glass rods encased in glass containers. Curve II: Bare (i.e. encased in onion-skin paper) glass rods.

The glass holders were of the same chemical composition as the rods to give equilibrium for induced β-ray emission from the activated glass elements. Fig. 1 shows the marked difference between their responses. The data for the glass-covered rods are given in Table 1, col. 2.

14-MeV neutron exposure

Rods encased in a lucite holder of 3 mm wall thickness or covered with aluminum foil of 11

* On leave of absence from National Institute of Genetics, Misima, Japan, to March 1959.

† Operated by Union Carbide Corporation for the United States Atomic Energy Commission.

Table 1. *Comparison of silver-activated phosphate glass exposed to γ-rays and neutrons*

Radiation	Unit glass response produced by exposure to	1 rad in glass is produced by (calculated)	Observed glass response per rad (arbitrary units)	Relative sensitivity of glass rod
Co^{60} γ-rays	7.35 r ± 5%*	1.18 r	0.161	1.00
Thermal neutrons	(1.84 ± 0.18) × 10^{10} (neutrons/cm²)* (2.06 × 10^{10})†	3.53 × 10^{9} neutrons/cm²	0.192 (0.171)†	1.19 (1.06)†
14-MeV neutrons	(1.73 ± 0.26) × 10^{10} (neutrons/cm²)*	$<10^{9}$ neutrons/cm²	<0.058 (0.033)†	<0.36 (0.21)†
	(3.0 × 10^{10})†	>5 × 10^{8} neutrons/cm²	>0.029 (0.017)†	>0.18 (0.11)†

* Overall measurement errors.
† Corrected values by taking account of γ-contamination.

mg/cm² were exposed to mono-energetic 14-MeV neutrons produced by $H^3(d, n)He^4$ reaction available in the Cockcroft–Walton accelerator in the Biology Division of Oak Ridge National Laboratory. The neutron flux was determined as the average value

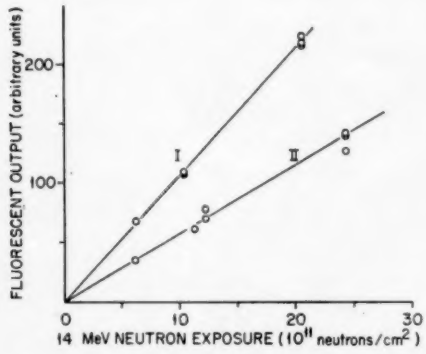

FIG. 2. Fluorescent output of silver-activated phosphate glass vs. 14 MeV neutron exposure. Fluorometer sensitivity: same as for Fig. 1. Neutron exposure rate range: 4–2 × 10^{8} neutrons/cm² sec. Curve I: Glass rods encased in a lucite holder. Curve II: glass rods covered by aluminum foil.

from a calibrated long-counter counting the number of neutrons and from an α-counter measuring directly the α-particles produced in 1 : 1 coincidence with the neutrons by the foregoing reaction. The measurement error in the neutron flux is less than ±10 per cent, and γ-contamination was (3 ± 2) per cent in terms of tissue rad.[9,10] The results are shown in Fig. 2. The 86 per cent higher response of lucite-covered rods compared with the aluminum-covered ones is attributable to recoil protons from the lucite holder because the amount of energy transfer from aluminum foil to glass rods through particles produced by the (Al, *n*) nuclear reactions is negligible. If we take into account the 3 per cent, in terms of rads in lucite, γ-contamination, the ratio of the response of lucite-covered rods to that of aluminum-covered ones becomes 2 : 1. The data in column 2, Table 1, are for aluminum-covered rods.

DISCUSSION

To compare neutron response of glass rods with γ-ray response, we should use the term "response per rad in glass" or its equivalent. Suppose only Compton effects are responsible for energy absorbed by glass for Co^{60} γ-rays. Then the value 1r = 87.7 ergs/g of air[11] can easily be converted into rads in glass by multiplying the ratio of the number of electrons per gram of glass and of air,[12] column 3, Table 1. The calculation is based on the elemental composition* of the glass: $Al(PO_3)_3$, 46.3 per cent; $Ba(PO_3)_2$, 23.15 per cent; KPO_3, 23.15 per cent; and $AgPO_3$, 7.4 per cent. The exposed rods were encased in lucite tubes of 3 mm thickness for secondary electron equilibrium.

* G. E. BLAIR (Bausch & Lomb Optical Co.), private communication.

The energy, D, imparted by mono-energetic neutrons to unit mass of the glass is[11,13]:

$$D = \Phi \sum_{k,l} n_k \sigma_{kl} E_{kl} \qquad (1)$$

where Φ is the number of neutrons per square centimetre, n_k the number of k nuclei per gram, σ_{kl} the cross-section for any interaction (l) involving nucleus k, and E_{kl} the total energy imparted as a result of this interaction to secondary radiations. We can easily find the number of thermal neutrons necessary to produce 1 rad in the glass (column 3, Table 1) by using activation cross-sections for thermal neutrons,[14] neglecting γ-ray contribution to E_{kl}, and using mean energy, $\overline{E_{kl}}$, of β-rays for E_{kl},[11] from the elemental composition of the glass just mentioned and equation (1). Rigorous calculation of equation (1) for 14 MeV neutrons is not possible because of the lack of the $\sigma(n, p)$ and/or the $\sigma(n, \alpha)$ cross-sections for high atomic number elements of the glass. But, using RANDOLPH's value[13] for oxygen nuclei and neglecting the contribution from all other nuclei, we can find an upper limit for the number of neutrons necessary to produce 1 rad in the glass. For the lower limit calculation, we may assume that the contribution of each nucleus, other than oxygen, is twice as much as that of the oxygen nucleus.* The results are given in column 3, Table 1. We obtain the fluorescent output of the glass rods per rad by dividing the values in column 3 by the corresponding values in column 2. Then, taking Co^{60} γ-ray response as unity, we have the relative sensitivity of glass rods as given in column 5.

More than 90 per cent of D given by equation (1) is caused by activation of Ag^{107} and Ag^{109} which gives rise to β-rays of mean energy about 1.2 MeV, and hence an appreciable amount of the energy escapes from the tiny glass rods; however, 3 mm thick glass walls are thick enough to give equilibrium for those β-rays.† This accounts for the different response of the bare and glass-covered rods shown in Fig. 1. From it we may conclude that about 65 per cent of the energy escapes. Because of the small difference of linear energy transfer (LET) values between $\overline{E_\beta} = 1.2$ MeV and the average energy of secondary electrons produced by Co^{60} γ-rays, the relative sensitivity for thermal neutrons is expected to be very close to unity. Then the fraction 0.19 of the apparent relative sensitivity given in Table 1 is attributable to gamma contamination in the thermal column. The average neutron flux was 6.3×10^8 neutrons/cm^2 sec. Then, assuming that the γ-contamination was 100 reps/hr[8], that 1 rep equals 1.075 r, and that the response of a rod per rad to the β-rays from activated-silver nuclei is equal to the response to secondary electrons produced by the γ-rays, we find the corrected relative sensitivity comes down to 1.06. Considering the foregoing assumptions, the measurement errors and ± 10 per cent uncertainty in cross-sections of silver nuclei,[14] the agreement is good.

Similar correction for γ-ray contamination has been applied to obtain the corrected response to 14-MeV neutrons. As seen from RANDOLPH's table,[13] the energy imparted to the exposed material by (O, n) reactions is almost all attributable to the heavy particles p, α and O. Then the lower sensitivity of the glass to 14-MeV neutrons means that the glass dosimeter reads less for high LET particles. This is very important if we are to compare the readings of glass rods exposed to different kinds of radiation. Once the sensitivity vs. LET curve is obtained, the absolute dosimetry of any kind of radiation will become possible. Because of the lack of established stable fluorescent standards, we recommend standardization of measurements by reference to samples irradiated in known Co^{60} γ-ray fields. Then meaningful comparisons can be made between readings of different workers at different places and times.

CONCLUSIONS

The fluorescent response of silver-activated phosphate glass per rad is the same for thermal neutrons as for Co^{60} γ-rays. More than 90 per cent of the absorbed dose and the response

* Thanks are due Dr. M. L. RANDOLPH for suggestion of this calculation.

† The inclusion of the β-ray energies from the silver isotopes as absorbed dose from slow neutrons, although perhaps questionable conceptually, seems admissible for these short-lived isotopes, but may not be satisfactory for long-lived isotopes.

to thermal neutrons is caused by the β-particles emitted from activated-silver nuclei. Once such glass is calibrated by Co^{60} γ-rays, it provides the means for determination of either thermal neutron fluxes or of γ-contamination in a thermal neutron field, if one or the other is known. The response of this glass to 14-MeV neutrons is only about 20 per cent of that to Co^{60} γ-rays producing the same absorbed dose in the glass and hence decreases markedly with increased LET. The 14-MeV neutron response for glass rods surrounded by a hydrogen-containing material is twice the response of rods wrapped in thin aluminum foil. This increase is believed to be caused by recoil protons and hence to be independent of γ-contamination. Such comparison methods may make the use of these glass rods both for neutron and γ-ray measurements in the same field possible. As shown here, the glass rods are primarily neutron-flux detectors, but given the chemical composition of any material, e.g. soft tissue, the abosrbed dose in that material may, in principle, be calculated by using cross-section and energy-transfer data.

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A RAPID INDICATING INSTRUMENT FOR THE STEPWISE MEASUREMENT OF AIRDUST RADIOACTIVITY

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Abstract—This paper describes some operational experience acquired with an instrument for measuring stepwise the radioactive contamination of air-borne dust. To bring down the background due to the natural radioactive gases always present in air, the α- and β-activity of the dust deposit are measured during the accumulation period. As there is a fairly linear relation between the activities due to the natural radioactive gases, it is possible by means of a simple circuit to balance the α- and β-count-rates such that a zero reading is obtained if only natural radioactive gases are present, and a positive or negative reading is shown for a real β- or α-contamination, respectively. With a circuit so adjusted it proves to be possible with an air flow of 4 m^3/hr and a sample time of 12 hr to demonstrate during the accumulation period current normal levels of atomic bomb fallout (10^{-12}–10^{-11} $\mu c/cm^3$).

Some discussion is devoted to the promising possibilities of using this instrument in plutonium laboratories to detect the MPC within several hours.

INTRODUCTION

MODERN views on public health in connection with atomic energy require continuous recording of the radioactive contamination of the air.[1] The method used for measuring the contamination of air dust depends largely on the situation. The main factors are: the kind of contamination and the relevant maximum permissible concentration, the amount of air to be sampled, the size of the sample, the counting efficiency and background of the detectors, the dust content of the air, the self-absorption in the deposit on the filter, the collection efficiency of the filter paper and the time required to get information on a possible contamination.

Minimum requirements for routine measurements of air-borne radioactivity have been discussed and specified by BLOK[1]. He finds that a minimum detectable level of 10^{-12} $\mu c/cm^3$ of air is acceptable for a continuous monitoring system. His calculations show that with an air-flow of 35 l/min through a good filter a maximum background count-rate of about 500 counts/min will be caused by the daughter products of natural radon and thoron. An alarm setting at 1000 counts/min will then indicate with a delay of 43 min a contamination level of 10^{-9} $\mu c/cm^3$ if the natural radioactive gases give rise to 500 counts/min; in case the natural background is negligible 86 min is required to pass the 1000 counts/min level. Three or four days after sampling when the daughter products of natural radioactive gases have decayed sufficiently, more accurate measurements of a possible contamination will be made. An instrument satisfying these requirements would be attractive for general monitoring of air-borne activity since it can be constructed simply and cheaply, provided one is disposed to change and measure filters every day.

For continuous monitoring of air dust several types of instruments have been constructed. They normally have a slowly moving strip of filter paper passing along a suction head in such a way that a continuous strip of dust or dust spots is obtained. The activity of this deposit is measured during or shortly after accumulation and also with a delay of some days after the period of sampling. These instruments can also be provided with alarms; however, they do not

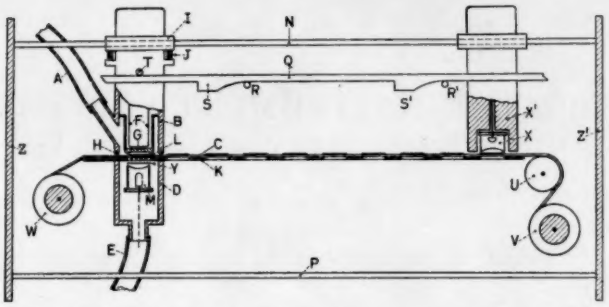

FIG. 1. Diagram of a continuously operating instrument for stepwise measurement of air-borne dust. The air enters the instrument through the hose *A*, passes the filter paper *C* and leaves through *E*. The activity of the deposit *L* is measured by means of an α-detector *F*, *G* and a Geiger–Müller counter tube *M*. The activities of samples are measured again after a delay of about 3 days by the Geiger–Müller tube *X*. (More details in text.)

give a warning essentially faster than the method discussed by BLOK (about 1 hr for 10^{-9} $\mu c/cm^3$).

It might be very important to detect even a slight contamination as soon as possible to provide a warning in the event of accidents. This paper gives some results acquired with an instrument, described in more detail elsewhere.[2] It operates with a filter-paper band moved stepwise over a suction grid (for constructional details see next paragraph and Figs. 1 and 2). During the dust accumulation the α-activity is measured by means of a detector immediately above the filter paper, while the β-activity is estimated with a β-detector underneath the suction grid. So both detectors are mounted in the suction circuit.

This arrangement makes a special background compensation possible as experiments and theoretical considerations prove a fairly constant ratio between the α- and β-activity of the background due to the natural radioactive gases in air. This holds also during the dust accumulation.[2] Suppose that the reading of the count-rate meter connected to the α-detector

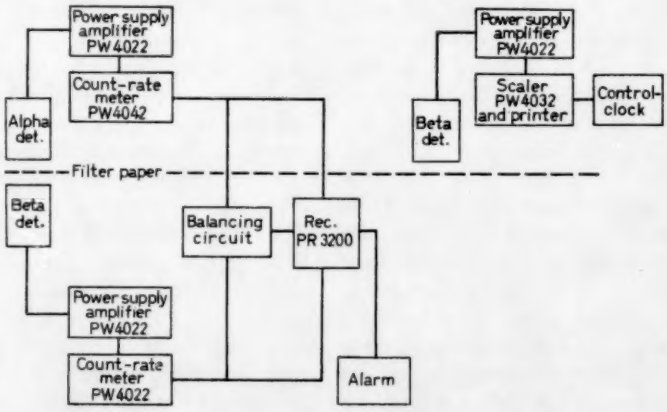

FIG. 2. Block-diagram showing the electronic units.

is A and that of the β-detector after correction for its inherent background, B. By virtue of the constant ratio, the signal C obtained by subtracting the reading A of the α-detector after multiplication with a constant factor k, from the corrected reading B of the β-detector (thus $C = B - kA$) can be made equal to zero by proper choice of k. Under this condition a β-contamination will cause a positive signal for $C = B - kA$ while an α-contamination will result in a negative output signal. A simple circuit performs this operation and so gives a positive reading when a real β-contamination is present and a negative reading when a real α-contamination is present.

Studies of the α- and β-activity of the daughter products of air containing a thoron concentration of 5 per cent reveal that during a suction period of 1 hr the β to α ratio is effectively set by the radon daughters. For a suction period of 10 hr this ratio is largely set by the thoron daughters. The k value in the equation $C = B - kA$ is for thoron about 25 per cent higher than for radon. Now, ThB emits a very soft β-radiation that can readily be absorbed by a thin foil. This makes it possible to equalize the ratio k for thoron and for radon by placing a foil of proper thickness between the source and β-detector. In this way equal values for the counted β to α ratio for the daughter products of radon and thoron can be obtained both during accumulation and thereafter, so that ignorance of the thoron percentage is no longer a hindrance to the background compensation.

Experience shows that one is able to detect a disturbance of the equilibrium if it exceeds about 10 per cent. So the sensitivity of this method is of the order of 10 times better than one without a compensation system in case the maximum concentration of natural radioactive gases is present. As in practice this concentration is generally lower and may vary by more than a factor of 10, the sensitivity of the method described here will be in most cases still more favorable.

In the next paragraph also will be elucidated the precise measurements of the activity of the dust samples a few days after accumulation, when the natural activity has decayed sufficiently.

INSTRUMENT AND MEASURING METHOD

The principle of the construction of the instrument is elucidated by the diagram given in Fig. 1. Through a hose A the air to be investigated enters the inlet tube B and is sucked through the filter paper C. It leaves the instrument through the outlet tube D and a second hose E. In the inlet above the place where the dust is collected, an α-counter is mounted consisting of a photomultiplier tube F with an acrylic resin light-guide G on which there is a thin layer of ZnS(Ag) crystals. The phosphor is not covered with a light shield and the distance between the phosphor and the deposit is only a few millimeters, which ensures optimum counting efficiency. The inlet tube is provided with a rubber ring H and can move in a vertical direction in the holder I. The springs J press the inlet tube to the support K of the filter paper, thus forming an air-tight and light-tight joint. The filter paper is supported by a grid L. Underneath the filter paper an end-window Geiger–Müller counting tube M provided with an equalizing foil Y is arranged.

Before transporting the filter paper, pumping is first stopped and the high tension of the photomultiplier tube removed. A few seconds later the inlet tube B is lifted by moving the bar Q to the right so that the ridges S and S^1 slip over the fixed points R and R^1 and raise the supports T and with them B. The filter paper is then moved over a fixed distance by means of the rubber roll U and wound on a roll V. Thereupon B is lowered, the high tension for the photomultiplier is again applied and pumping started. The paper is supplied by a roll W. The two bars N and the two bars P are fixed to the plates Z and Z^1, thus forming a rigid frame. The transport of the paper and the switching of the several operations is effected by one motor mounted on Z^1. A clock giving signals at adjustable fixed times controls the motor. For the gas circuit a rotary pump is used. The pump can be shunted by an overflow valve to ensure a constant flow.

The above-mentioned paper[2] describes a second set of detectors for α- and β-counting designed to measure the contamination according to the same compensation method. As fallout activities are normally only measured some days after the accumulation and mainly only checked for β-emitters, an arrangement is used here where the samples are measured with a delay of some days by a β-counter X supported in a lead shield X^1. The counts of this β-counter are recorded by a printer controlled by the clock. A block diagram of the various units is given in Fig. 2. The type numbers refer to commercial Philips products.

The instrument was adjusted for sampling times of

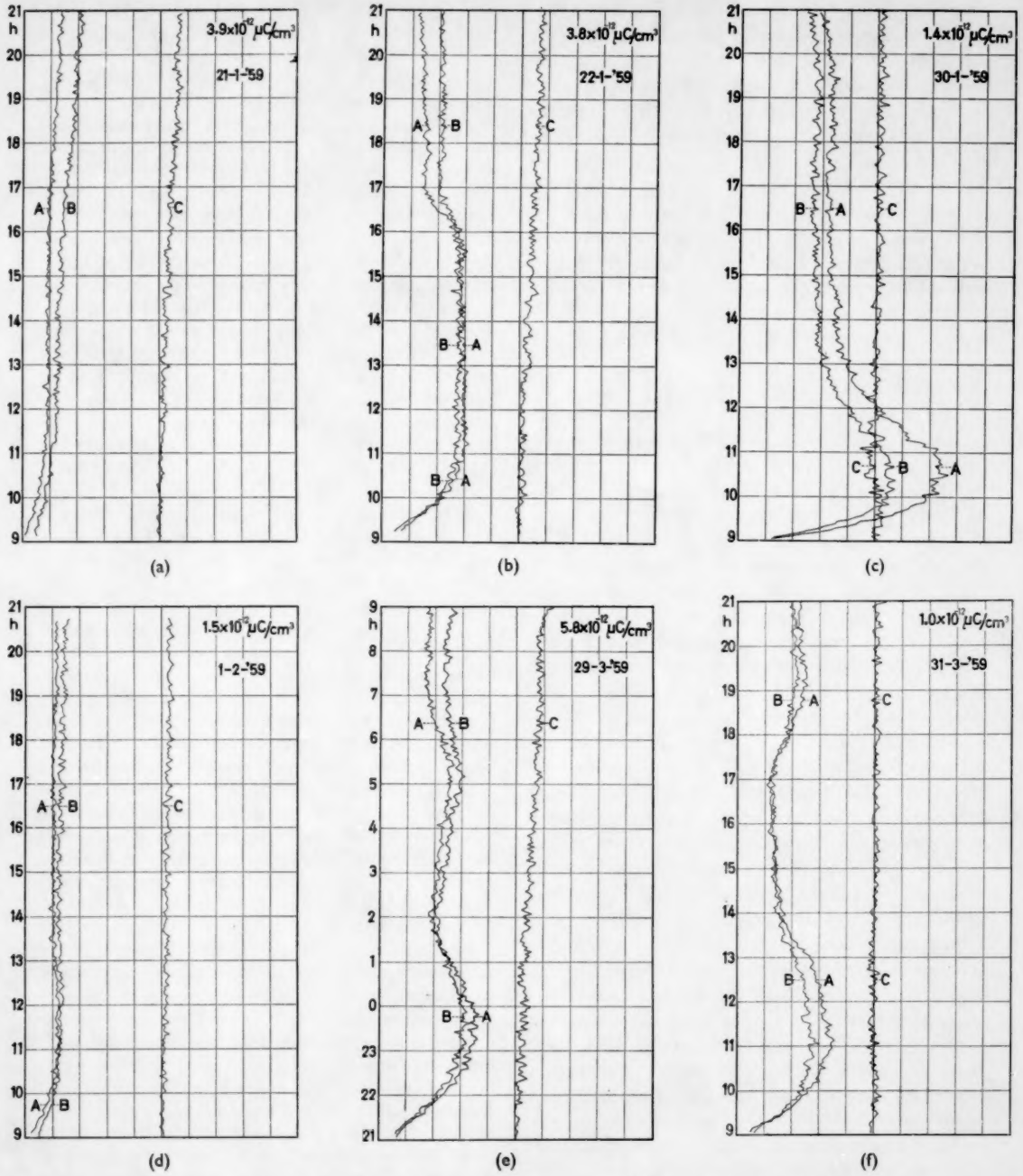

FIG. 3. Some typical strip charts. The reading of the α-count-rate meter is indicated by **A** and that of the β-count-rate meter (not corrected for the background of the Geiger–Müller tube) by **B**. The effect of the balancing circuit is denoted by **C**. In the upper right of the charts the contamination level is given as has been measured after about 3 days. At the ordinate, time is given in hours. For the **A** and **B** curves the abscissa ranges from 0 to 1000 counts/min. In case of a β-contamination the **C** curve runs to the right from 0 up to maximal 500 counts/min. By an α-contamination the **C** curve can run to the left from 0 to a maximum of 500 counts/min.

12 hr so that two samples a day were obtained. The flow-rate amounted to about 70 l/min, each sample corresponding to the air dust content of about 50 m^3 of air.

The filter paper is made of micron glass fibres, type 1106-B, manufactured by the Mine Safety Appliances Company of Canada. The paper is used in the form of rolls 3 in. wide. The effective area of the suction grid amounts to 8 cm^2 and with a flow of 4.2 m^3/hr the pressure drop over the paper is 10 cm Hg (face velocity of air, 1.5 m/sec). The efficiency of this type of paper has been studied by FITZGERALD and DETWILER[3]. It amounts to 99.9 per cent for particles with a diameter larger than 0.04 μ and has a dip (93 per cent) for particles of 0.01 μ (face velocity 10 cm/sec). Higher face velocities give still better efficiencies.

The count-rate meters were used with an RC time of 1.67 min. The instrument has been run completely automatically for 5 months, the only failure being caused by minor pump troubles.

EXPERIMENTAL RESULTS

As mentioned in the preceding paragraph three signals are fed into the recorder: the count-rate of the α-detector, the count-rate of the β-detector and the result of the balancing circuit. Fig. 3 gives some typical examples of these strip-charts. The reading A of the α-detector is indicated by curve **A**, the reading of the β-detector is shown by curve **B** and is not corrected for the background. After this correction the value B can be obtained. The effect of the balancing circuit which provides the signal $C = B - kA$ is denoted by curve **C**. It should be noted that zero values for A and B are at the left side of the record while $C = 0$ corresponds to a line in the centre of the chart.

Each strip-chart covers a period of 12 hr. The numbers on the left-hand scale refer to the hours of the day, the curves **A** and **B** demonstrate the build-up of the natural radioactivity during the first 2 hr. Curve **C** shows the build-up of the real contamination of the air.

The graphs for **C** in case of various contamination levels are illustrated in Fig. 4. These curves have been calculated on the basis of the actual situation (flow rate 70 l/min, total counting efficiency of the β-counter underneath the filter paper 18 per cent, range of **C** between −500 and +500 counts/min).

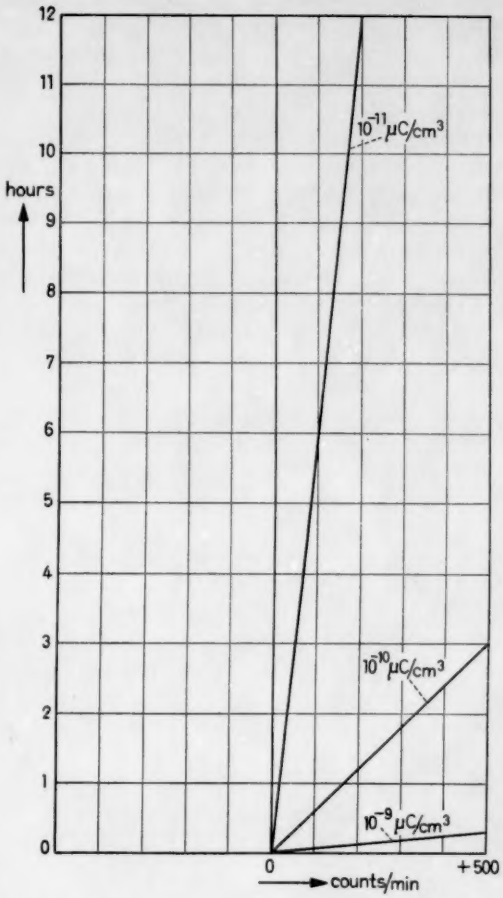

FIG. 4. Idealized graph for **C** curves in respect of the indicated contamination levels.

The calculated curve for a contamination level of 10^{-11} μc/cm³ fits the experimental results shown in Fig. 3, where the mean air contamination over a period of 12 hr was measured more precisely 3 days after accumulation. These values are marked in the upper right of the corresponding records.

The data of Fig. 5 give a survey of daily maximum and minimum β-background levels in counts per minute as obtained from the strip-charts [curves (a) and (b)]. The minimum values were estimated at least 2 hr after the start of a new sample period in order to avoid the build-up period. In the same figure the air

contamination is shown as measured by the balancing circuit [curve (c)] and as estimated about 3 days later [curve (d)]. To facilitate a comparison with the natural background, the curves (a), (b) and (c) are given in counts per minute. For curve (d), however, the units $\mu c/cm^3$ were used. It should be noted that during these experiments the air flow was not stabilized, introducing an error of about 25 per cent in the total amount of sampled air partly due to changes in the dust content of the air and the inherent variation in the flow resistance of the filter paper. This results in the same error of the absolute values as given by (c) and (d). It does not interfere, however, with the ratios of the corresponding values from (c) and (d). It should also be mentioned that weather conditions during January 1959 were sufficiently varied in temperature, humidity, etc., to give a fair example.

CONCLUSIONS

From the results presented in the foregoing paragraph it is clear that the compensation method is a practicable one even in industrial towns such as Eindhoven with a substantial amount of industry and traffic dust. It may be mentioned that the sampling place is situated at a distance of 50 m from quite a busy road and at the border of a factory complex of 0.3 km^2 where more than ten thousand workpeople are employed. The mouth of the sampling tube was mounted, facing downwards, at a height of about 10 m.

From Fig. 5 it is clear that an alarm setting for $C = 100$ counts/min might be practical. As can be seen from Fig. 4 this implies a warning signal within, respectively, 3.6, 36 and 360 min for β-contamination levels of 10^{-9}, 10^{-10} and 10^{-11} $\mu c/cm^3$. With shorter sampling times a lower alarm setting is justified and shorter warning times are obtained. The next step will be to use these instruments in a plutonium laboratory. The MPC for plutonium in air amounts to 2×10^{-12} $\mu c/cm^3$ (Table C VIII of Ref. 4). This value is based on continuous exposure and can be multiplied by a factor of 3 for usual occupational exposure (see Ref. 4 part C, chapter X, paragraph 9). From Fig. 4 one can deduce that an α-contamination of 6×10^{-12} $\mu c/cm^3$ will give rise to a negative reading from C of 120 counts/min after a sampling time of 12 hr. This presupposes that the air is sufficiently filtered from any atom

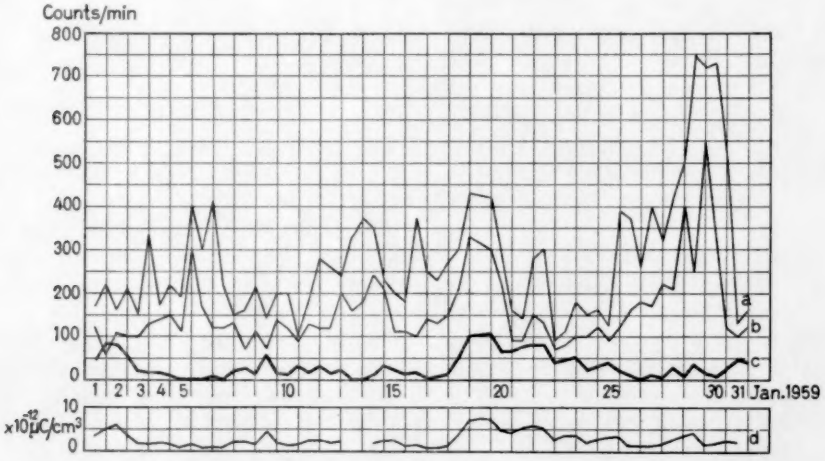

FIG. 5. Survey of the measuring results over January 1959. Curves (a) and (b) give the maximum and minimum values of the natural background in counts per minute. Curve (c) summarizes the contamination levels (in counts per minute) as can be deduced from the **C** curves (see Fig. 3). Curve (d) shows the contamination levels in microcuries per cubic centimeter as measured after a delay of about 3 days.

bomb fallout. These filters will also remove the dust bearing the daughter products of natural radioactive gases. Only radon and thoron will pass through the filters. Artificial contamination of the air by radon and thoron demonstrated that one may expect that the method described in this paper will also be applicable in rooms with highly filtered air. In order to raise the sensitivity, the air flow can be increased and it will be possible to detect the MPC within several hours. Owing to the low dust content of the air it will also be possible to take samples over periods of several days, thus accumulating more contamination dust. To make more accurate measurements when the natural radioactivity has sufficiently decayed the instrument will be provided with a second α-detector.

The degree of accuracy of the rapid indication is set by the constancy of k and by counting statistics. In our experiments the second factor dominates. The counting statistics depend on the time constant of the integrating network of the count-rate meter. Although this constant was not optimalized one gets the impression that the chosen value of 1.67 min gives a practical compromise under the rather variable circumstances.

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LATENT-IMAGE FADING IN FILM BADGE DOSIMETERS

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Abstract—The primary purpose of this study was to establish the feasibility of using a cumulative film as well as a weekly film in film badge dosimeters, since use of a 3 month cumulative film would allow about a tenfold reduction in minimum reported dose over that obtainable with weekly film. The parameter of most interest is the film latent-image fading period. This was investigated for times up to 90 days under varying environmental conditions. The controlling factor was found to be humidity. Use of a film wrapper impervious to water vapor should effectively solve the problem.

INTRODUCTION

As EARLY as 1910 the phenomenon of photographic latent-image growth and fading was observed and studied.[1] Information regarding this effect is of considerable importance in the field of film badge dosimetry where reliable interpretation of the photographic density of films subjected to varying humidity and temperature over long latent-image fading periods is often required. Latent-image fading for storage periods up to 8 days has been reported.[2] However, no data covering longer periods of time and the effect of various humidities and temperatures have as yet been published. The primary purpose of this study was to establish the feasibility of using a "cumulative" film as well as a weekly film in film badge dosimeters. A cumulative film developed every 3 months would provide a more than tenfold reduction of the minimum reported exposure dose over that obtained by numerical addition of the minimum detectable doses of films processed on a weekly basis. It is possible to utilize this gain in sensitivity only if latent-image fading can safely be neglected.

LATENT-IMAGE FADING

While the mechanism of latent-image fading is not completely understood, it is generally agreed that it is primarily a chemical process controlled by the atmosphere surrounding the film. Humidity and to a lesser extent atmospheric oxygen content are known to be influencing factors. Also, at least a small part of residual fading of particle tracks in nuclear emulsion has been attributed to thermal effects.[3] The present tests were carried out using only one type of dosimeter film packet, i.e. du Pont packet type 552 containing film types 502 and 510. This packet was chosen because it is widely used for radiation monitoring and because preliminary data indicated that its component films should have good latent-image fading characteristics. Since previously published data have established the growth and fading pattern of these films over a weekly period, it was not felt necessary to deal with fading periods less than 7 days in the present work. Because in practice film dosimeters are subject to varying temperatures it was decided to conduct fading experiments using three representative values, 27°, 34° and 43°C, covering the range from moderate to rather high atmospheric temperatures. In order qualitatively to determine the long range effects of oxygen and humidity on image fading, films were stored in two atmospheres, room air and dry argon. No attempt was made to control the humidity of the room air storage since it was desired to determine the behavior of film under practical operating conditions. Finally, to determine whether or not the magnitude of the exposure dose of ionizing radiation to which the films were exposed influenced fading, all tests were conducted using five roentgen values, 0.043, 0.173, 0.405, 1.190 and 2.380.

Exposures were made using γ-rays from cobalt-60. The variation in exposure dose to any given film batch did not exceed ± 2 per cent. After exposure, films were placed in temperature-controlled atmospheres of either dry argon or room air. The temperature variation during storage was ± 2°C.

All films were taken from the same emulsion batch and processed together after storage. Their diffuse

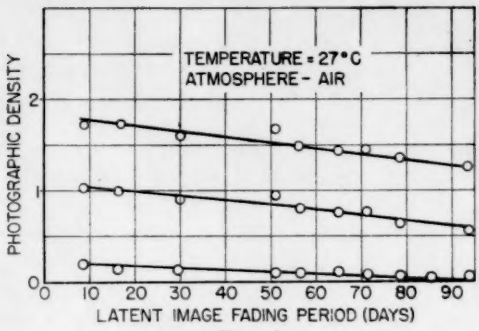

FIG. 1.

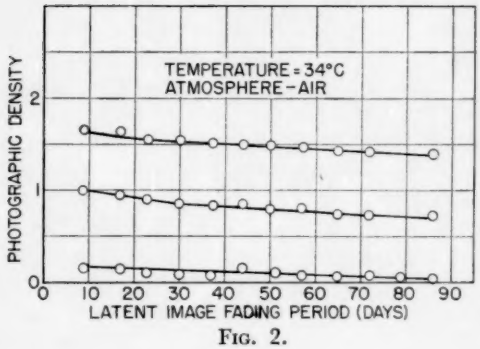

FIG. 2.

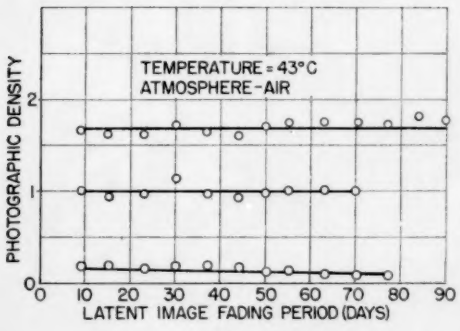

FIG. 3.

FIGS. 1–3. Latent-image fading of films stored in room air (uncontrolled humidity) at three temperatures. Results for three exposure doses are shown.

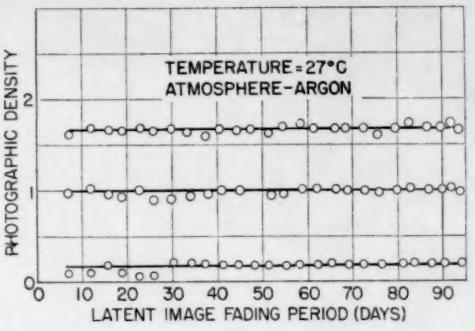

FIG. 4.

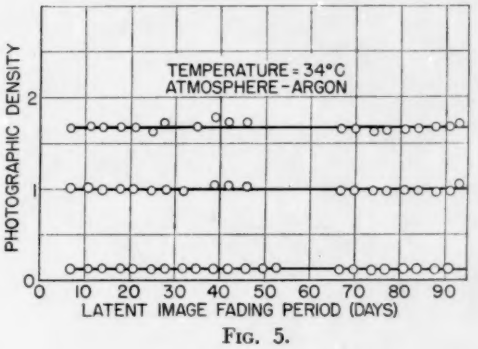

FIG. 5.

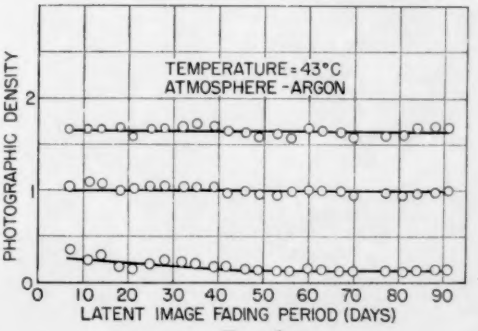

FIG. 6.

FIGS. 4–6. Latent-image fading of films stored in dry argon at three temperatures. Results for three exposure doses are shown.

transmission densities were measured by means of a standard type of photoelectric densitometer. In processing, Kodak Liquid X-Ray Developer was used for 5 min at 20°C.

RESULTS

Photographic density vs. latent-image fading period, i.e. the time elapsed between exposure and processing, is plotted in Figs. 1–6. The type of atmosphere and its temperature in which films were stored are indicated on the figures. Three densities corresponding to 2.380, 1.190 and 0.173 r are shown as being representative of the type of data obtained. Only data for the sensitive film of the 552 packet are shown, since information concerning the low exposure dose region is of primary importance in establishing the feasibility of the cumulative film badge dosimeter. It was found, however, that the fading characteristics of the insensitive film were essentially the same.

The results indicate that the overall stability of this film is excellent. Temperature, in the range up to 43°C, is a negligible factor in the latent-image fading of this film. Moreover, the effect of the presence of atmospheric oxygen is also negligible. Both of these conclusions can be drawn from the curves taken at 43°C in room air and in argon, where in neither case was detectable latent-image fading found to occur. The controlling factor is evidently humidity. This is indicated by Figs. 1 and 2 where appreciable latent-image fading for films stored in room air at lower temperatures occurred. The films exposed at 27°C show a greater fading than those stored at 34°C, while those stored at 43° in air show no detectable fading. Since it is well known that lower temperatures inhibit fading (and none of the films stored in argon at the same lower temperatures show detectable fading), this cannot be a thermal effect. However, it is simple to calculate (since these films were stored in three containers, all providing access to room air but maintained at three different temperatures) that films stored at 43°C were subject to an atmosphere having on the average about one third the relative humidity as those stored at 27°C (if we assume an average relative humidity at 27°C of 50 per cent). These data show, therefore, that the relative humidity normally encountered by a film badge dosimeter is enough seriously to impair its reliability as a dosimeter over long storage periods. For example, Fig. 1 indicates that an exposure dose of 1.190 r would be interpreted lower by a factor of about two after 90 days storage in "normal" room air (during the 90-day storage period the room air relative humidity fluctuated between 20 and 80 per cent). However, it is a relatively simple matter to protect the film with a wrapper that is impervious to water vapor. The results of these experiments indicate that the use of the 552 packet as a cumulative film dosimeter should be entirely feasible, if it is suitably packaged.

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SHORTER-LIVED FISSION PRODUCTS IN FALLOUT

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Abstract—Calculations are made of the contribution of shorter-lived fission products in radioactive fallout to the worldwide external γ-levels. These are related to potential cesium-137 radiation doses. Certain assumptions are necessarily used in the calculations, one of which is the mean residence time in the atmosphere. The potential theoretical dose from shorter-lived isotopes may vary from about 40 per cent (mean or average life in the atmosphere, $T_m = 10$ years) to 1000 per cent ($T_m = 2$ weeks) of the cesium-137 dose. Estimates are also made of the worldwide total deposition of cesium-137 from which calculations are made of the potential cesium-137 doses.

THE contribution of shorter-lived fission products to the external γ-levels of worldwide radioactive fallout is not well documented. There are a few isolated data available[1–3] and estimates have been made as to their relative dose *rates* (to cesium-137) at the time of measurement.[1] However, the measured activities of the shorter-lived isotopes represent a mixture of fallout debris from many nuclear detonations in the past, falling in varying amounts, after differing storage times in the stratosphere and troposphere. Any quantitative estimates of total radiation *doses* from the time of deposition based on the measurements of current amounts of shorter-lived isotopes would be most tenuous. Since it is not now possible to make additional and sufficient direct measurements (too long a time has expired since the last nuclear test detonation) this paper attempts to make estimates of relative total doses (short-lived fission products vs. cesium-137) by a theoretical approach.

METHODS EMPLOYED

This approach is to estimate the total potential γ-release at the earth's surface from the short-lived isotopes relative to cesium-137. The procedures used are: (a) tabulate all pertinent radioisotopes in mixed fission products using slow neutron fission data;[4] (b) calculate the relative total potential γ-energy releases based on the relative abundances of atoms and their γ-energy releases per disintegration;[4–6] (c) assuming half-times of residence in the troposphere and stratosphere, calculate the potential γ-energy release at the earth's surface.

Table 1 presents the isotopes of principal concern and the method used to obtain relative (to cesium-137) potential γ-energy releases from these short-lived fission products. There are, of course, many additional short-lived isotopes in fresh mixed fission products but these account for less than 10 per cent of the total potential γ-energy release at the earth's surface outside of local areas around the points of detonations, owing to their short half-lives, low abundance yields and/or low energy releases per disintegration.

Table 2 illustrates the method used to arrive at the theoretical relative (to cesium-137) γ-energy releases on the earth's surface.

DISCUSSION

There are three principal factors related to the calculated results shown in Table 2. One, the longer the fission products remain in the atmosphere the less percentage of total potential γ-energy release that occurs on the earth's surface. The mean (average) residence time may vary for different altitudes of the troposphere[7] but a range of from 2 to 4 weeks is representative of several estimates.[8] Likewise, the mean residence time in the stratosphere may not be a single value (as indeed the whole

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Table 1. Potential γ-energy releases (relative to cesium-137) from short-lived fission products

Isotopes	Half-life	γ-energy (Avg. MeV/dis.)	Number of atoms per 10^4 fission (at about 24 hr)	Total potential energy release col. 3 × col. 4	Relative (to Cs^{137}) potential energy release
(Te^{137}—I^{137}—Xe^{137})—Cs^{137}(Ba^{137m})	27 years	0.602[5]	578	348	1.0
(I^{140}—Xe^{140}—Cs^{140})—Ba^{140}—La^{140}	12.8 days	0.180	578	1500	4.32
	40 hr	2.42	26		
(Rb^{95}—Sr^{95}—Y^{95})—	65 days	0.719	623		
Zr^{95}→Nb_1^{95m} (3%)	90 hr	0.233	—	913	2.62
↘ ↓					
Nb_2^{95} (97%)	35 days	0.702	6.4		
(Sn^{132}—Sb^{132})—Te^{132}—I^{132}	77.7 hr	0.213	312	760	2.18
	2.4 hr	2.224	9.8		
(Zr^{103}—Nb^{103}—Mo^{103}—Tc^{103})—Ru^{103}—Rh^{103}	39.8 days	0.486	334	171	0.492
	57 months	0.027	—		
(La^{149}—Ce^{149}—P^{149}—Nd^{149})—Pm^{149}	47 hr	1.28	102	131	0.377
(Sr^{99}—Y^{99}—Zr^{99}—Nb^{99})—Mo^{99}—Tc^{99m} (13%)	68.3 hr	0.143	472	112	0.323
	5.9 hr	0.869	5.3		
(In^{131}—Sn^{131}—Sb^{131})—					
↗ Te_1^{131m} (5%)	30 hr	1.43	24.6		
↘ ↓					
Te_2^{131} (95%)—I^{131}	25 months	0.348	0.35		
	8.1 days	0.3906	263	113	0.325
(Cs^{144}—Ba^{144}—La^{144})—Ce^{144}—Pr^{144}	290 days	0.0323	499	64	0.184
	17.5 months	0.0949	—		
(Ba^{147}—La^{147}—Ce^{147}—Pr^{147})—Nd^{147}—Pm^{147}	11.6 days	0.1712	273	47	0.135
	3.7 years	No γ	17		
(Xe^{141}—Cs^{141}—Ba^{141}—La^{141})—Ce^{141}	33 days	0.0792	583	46	0.132
(Nb^{106}—Mo^{106}—Tc^{106})—Ru^{106}—Rh^{106}	1.0 years	No γ	49.9	15	0.043
	30.0 sec	0.304	—		
(La^{151}—Ce^{151}—Pr^{151}—Nd^{151})—Pm^{151}—Sm^{151}	27.5 hr	0.369	27.7		
	73 years	0.275	22.3	14	0.040

Table 2. Theoretical γ-energy releases (relative to cesium-137) from short-lived fission products on the earth's surface
(Assumed mean time in stratosphere of 1 year)

Isotope	$\frac{\lambda_d}{\lambda_d + \lambda_r}$*	Relative total (infinity) energy release	Relative energy release on ground col. 2 × col. 3	Col. 4 based on $Cs^{137} = 1$
Cs^{137}	$\frac{1}{1 + 0.0257}$	1	0.975	1.0
Zr^{95}—Nb^{95}		2.62	0.668†	0.685†
Ba^{140}—La^{140}	$\frac{1}{1 + 19.8}$	4.32	0.208	0.214
Ce^{144}—Pr^{144}	$\frac{1}{1 + 0.873}$	0.184	0.0983	0.101
Ru^{103}	$\frac{1}{1 + 6.36}$	0.492	0.067	0.0686
Te^{132}—I^{132}	$\frac{1}{1 + 78.3}$	2.14	0.0275	0.0282
Ru^{106}—Rh^{106}	$\frac{1}{1 + 0.693}$	0.043	0.0254	0.0261
Ce^{141}	$\frac{1}{1 + 7.67}$	0.132	0.0152	0.0156
Te^{131}—I^{131}	$\frac{1}{1 + 31.3}$	0.325	0.0101	0.0103
Nd^{147}—Pm^{147}	$\frac{1}{1 + 21.8}$	0.135	0.0059	0.0061
Mo^{99}—Tc^{99}	$\frac{1}{1 + 89}$	0.323	0.0036	0.0039
Pm^{149}	$\frac{1}{1 + 129}$	0.377	0.0029	0.0030
				1.1628
Relative energy release on earth's surface		$\frac{\text{short-lived}}{Cs^{137}}$		
30-year duration of release:		~2.2		

* This simplified approach introduces an insignificant error for cesium-137 when computing relative 30-year and 70-year energy releases. (λ_d, storage time constant; λ_r, radiological decay constant.)

† Includes gammas from Nb^{95} as well as Zr^{95}, calculated by the same method as for other isotopes.

concept of exponential removal from the stratosphere may be only an approximation). For example, the HASP (Department of Defense) program[9] estimates a stratospheric residence time of a year or less if a single value is chosen for mean times of from 13 to 35 months for low-latitude clouds and from 4 to 13 months for U.S.S.R. tests. Their preferred values are 8.5 months for U.S.S.R. and 17.5 months for U.S. and U.K. low-latitude stratospheric clouds. From first principles it is very likely that high-altitude clouds have a longer residence time than low-altitude clouds. Also, MARTELL estimated "... that the stratospheric storage time is from 6 months to 1 year for Soviet tests, 1 to 2 years for recent U.S. equatorial tests and 5 to 10 years for the large scale U.S. Castle test of the spring 1954."[2]

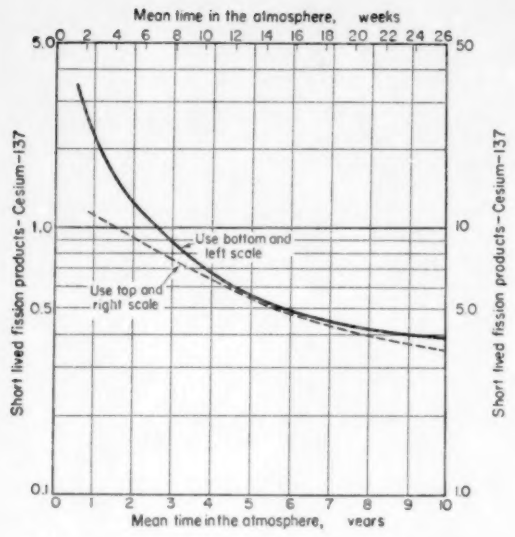
Mean time in the atmosphere, weeks
0 2 4 6 8 10 12 14 16 18 20 22 24 26
5.0
1.0
0.5
0.1
50
10
5.0
1.0
Short lived fission products - Cesium-137
Short lived fission products - Cesium-137
Use bottom and left scale
Use top and right scale
0 1 2 3 4 5 6 7 8 9 10
Mean time in the atmosphere, years

FIG. 1. Theoretical relative 30-year doses for (short-lived fission products)/cesium-137.

In light of the variances and uncertainties, calculations of theoretical relative γ-energy releases at the earth's surface from shorter-lived isotopes vs. cesium-137 were made for mean residence times ranging from 2 weeks to 10 years (see Fig. 1). The relative values were based on calculated 30-year exposures. For 70-year exposures, multiply the ratios by 0.64.

The second factor that influences the calculated results is the assumed percentage of fission products initially introduced into the stratosphere and troposphere at the time of a nuclear detonation. LIBBY[7] estimates 1 per cent of the fission products from high-yield tests to be tropospheric. Based on the gross fission product fallout, 35 days after the Castle Bravo event, MACHTA and LIST suggest that tropospheric fallout might be as high as 5 per cent. As a base line to estimate both (a) the relative tropospheric and stratospheric contribution to the fallout on the earth's surface, and (b) the total deposition of cesium-137 as a function of latitude, Fig. 2 was prepared. The stratospheric and tropospheric contributions have been estimated by (a) using the tropospheric data from MACHTA and LIST[10], and adjusting for additional contributions from the nuclear tests that have occurred since the spring 1958 until the end of tests in 1958, and (b) by using peak predicted levels on the ground of strontium-90 from reference (11), which were then corrected for decay. These values of total deposition of strontium-90 were increased by a factor of 1.8 to extrapolate to cesium-137. Measured values of cesium-137 vs. strontium-90 activity have varied depending upon the type of collection (air, rainfall, etc.) and place of collection (altitude and latitude).[12,13] The theoretical production ratio of cesium-137 to strontium-90 activity of about 1.8 was used here although it may be a little high.

Thirdly, the ratio of cesium-137 to shorter-lived fission products found in any one locality may show the effects of fractionation, i.e. the gaseous precursor xenon-137 (half-life 3.9 min) will tend to reduce the anticipated proportion

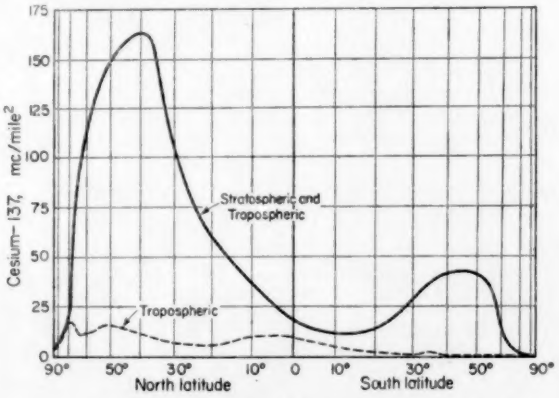

FIG. 2. Estimated total deposition of cesium-137.

of cesium-137 to shorter-lived fission products in nearby fallout from surface bursts and thus increase it in more distant fallout. This is because during the critical times when the temperature of the fireball and its contents (radioisotopes and inert materials) are optimum for incorporation of the fission products into the condensing particles, the gaseous precursor xenon-137 will continue to rise with the hot gases into the higher atmosphere. However, more than one-half of the precursor (to cesium-137) atoms, are initially nongaseous tellurium (half-life 3 sec) and iodine (half-life 22 sec) before they decay to xenon, allowing time for some incorporation in particles that fall nearby following surface bursts. Also, about one-half of the total fission yield from past nuclear tests have been air bursts where there is much less opportunity for fractionation. Further, more than one-half of the atoms that are precursors to barium-140–lanthanum-140 (a heavy contributor to the short-lived fission product γ-energy release for the first several weeks) also originate as gaseous xenon-140 (half-life of 16 sec). In light of these points and the fact that the limited data on relative abundances of cesium-137 vs. other fission products show substantial variability,[12] the decision was taken not to attempt to assign a quantitative value to possible fractionation.

Two additional factors should be mentioned. One, the actual fission product energy release to the air near the earth's surface may differ from the calculated ratio (shorter-lived fission products vs. cesium) owing to different degrees of weathering effects on the isotopes. In general, the total energy from the shorter-lived fission products will be released in much less time, thus providing less opportunity for weathering effects to occur. Secondly, the abundance of production of different isotopes varies somewhat from slow neutron fission (upon which these calculations are based) to fast neutron fission as would occur in a nuclear detonation.[14] The differences are not considered of major importance for the present problem.

ILLUSTRATIONS

From the two graphs, it may be seen that the relative (to cesium-137) potential γ-energy releases from short-lived isotopes may vary widely with latitude. Assuming, as an illustration only, a stratospheric mean time of 3 years and tropospheric mean time of 4 weeks, the theoretical relative total γ-energy release of short-lived isotopes to cesium-137 is about 1.5 over a 30-year exposure period at 40° N and about 5.5 at 0° latitude. However, the total cesium-137 *plus* short-lived γ-energy release at 0° latitude is only about three-tenths of that at 40° N. (Also, a distinction should be made between energy release *rates* and *total* energy releases. For example, at 2 weeks after a nuclear detonation the short-lived isotopes may produce nearly 4000 times the energy release *rate* as does the associated cesium-137.)

The estimation of actual radiation dose rates and doses to persons introduces still more uncertainties but it may be done, with necessary qualifications. A deposition of 1 mc/mile^2 of cesium-137 on an infinitely flat plane will result in about 0.03 mrad/year. Assuming a mean time in the stratosphere of 3 years (as an illustration) and no weathering effects, the peak activity at the areas of highest deposition (about 40° N) would be about 130 mc/mile^2 of cesium-137 occurring about 1964. This would produce a dose *rate* of 3.9 mrads/year, if the cesium-137 were spread over an infinitely flat plane and there had been no weathering or shielding effects.

Small additional cesium-137 fallout from the stratosphere after 1964 would increase the total deposition to about the equivalent of 138 mc/mile^2 or a total calculated theoretical maximum dose of 87 mr for the 30 years following this peak. Weathering and shielding factors have been assumed to reduce this by a factor of 10.[8] Thus, the 30-year exposure might be 8.7 mrads from cesium-137. With an assumed mean time of 3 years in the stratosphere and 4 weeks in the troposphere, and with the ratios of stratospheric and tropospheric fallout as shown in the graph for 40°N, the additional dose from short-lived isotopes would be about 130 mrads (theoretical maximum). The major portion of the total potential dose from the short-lived isotopes would occur within the first months and years after a detonation, while that from cesium-137 would be protracted,

which not only would result in a different distribution of radiation doses in time, but also the weathering effects would have less time for operation on the shorter-lived isotopes. In fact for the example above, essentially all of the potential radiation exposure from short-lived isotopes would be completed before the peak of cesium-137 activity occurred in 1964. Of course, the above example over-simplifies the problem since the cesium-137 and shorter-lived isotopes may have arrived at this latitude (40°N) after spending differing residence times in the atmosphere depending upon the location and types of nuclear detonations.[2] More detailed analyses can be made when more definitive information becomes known on the mechanisms of fallout.

Also, all areas at a given latitude will not receive the same amount of fallout. Differences in quantity of rainfall (which carries down the activity from the lower troposphere), normal variances in other meteorological factors, and the nearness of a given locality to the point of nuclear detonation (downwind), will result in unevenness in distribution. This may be illustrated by the fallout which occurred in South Dakota on 17 July 1957 following a nuclear detonation on 17 July 1957 at the Nevada Test Site. At the time the air mass from Nevada (containing the radioactive debris) passed over South Dakota, rainfall occurred resulting in a relatively high deposition of fallout. The theoretical infinity dose was estimated to be up to 550 mr in limited areas. Using the estimates of weathering and shielding effects made in reference 8, the actual dose might be about 55 mrads for this single event. Of course, higher values than these have been observed close to the sites of nuclear detonations, but these are considered "local" fallout and have been described elsewhere.[15–17]

CONCLUSIONS

The above estimates are not intended to be exact but they do indicate that the worldwide potential external γ-exposures from shorter-lived fission products in fallout can be significant compared to cesium-137. However, an acceptable degree of confidence cannot be established from current information. Further studies are needed to elucidate the mechanisms of fallout especially storage times in the atmosphere, distribution of the fallout material and the degree of fractionation of fission products. Further studies are also needed on the potential contributions of induced activities in fallout debris. Lastly, although most of the fission products in fallout debris are relatively insoluble, further studies should be made of internal emitters in addition to strontium-90 and cesium-137.

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THE ENERGY SPECTRUM OF γ-RADIATION FROM FALLOUT

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Abstract—This report compiles and analyzes the results of fallout sample gamma-ray spectral measurements using a NaI(Tl) crystal scintillation spectrometer. The summary of the information indicates that, for those detonations at which γ-ray spectral characteristics of deposited fallout have been measured, there appears to be no positive correlation between spectral characteristics and weapon yield, weapon type or conditions of detonation. Comparisons have been made between the experimental results and the calculational predictions both of NELMS and COOPER, and of DOLAN for the γ-radiation from the products of slow neutron fission of U^{235}. Although there is considerable scatter in the experimental data, agreement between experiment and calculations appears relatively good, except for variable effects produced by neutron-induced radioactivity, such as Np^{239}. The mean energy per photon in the time interval between 10 and 3000 hr after detonation appears to be not too different from 0.7 MeV/photon if some shielding is provided to screen out low-energy photon radiations.

INTRODUCTION

SOME form of radioactive fallout is observed whenever a nuclear weapon is exploded. The degree of concentration of fallout in the vicinity of the point of detonation (ground zero) is dependent on the distance above or below the surface of the earth at which the detonation occurs, and upon the yield of the weapon.[1] If a weapon is detonated near the surface of the earth, it is almost certain that there will be regions in the vicinity of the detonation in which there are lethal γ-ray radiation doses. The exact distribution of the radioactive fallout is dependent on the meteorological conditions at the time.[2]

In a near-surface burst, large quantities of earth and debris are taken into the fireball in its early stages. This material is fused or vaporized and becomes intimately mixed with the radioactive products that have been produced by the weapon. After cooling, all this material must settle to the earth's surface, and the large quantity of material that has been mixed with radioactive debris assists in bringing much of the radioactivity back to the earth within a few miles of ground zero.[1] On the other hand a weapon which is detonated at high altitude has only the materials of the weapon on which to condense. In this case the radioactive products are in the form of an exceedingly fine aerosol[2] and may be carried through the air for many hundreds of miles before finally diffusing to the earth's surface or being brought down by rain or other precipitation.[3] For any air burst at sufficient height that little or no matter is picked up from the earth's surface, the only residual radioactivity observed near ground zero is that which is induced in the soil or other materials in the vicinity by the neutrons from the weapon. In most soils the most significant neutron-induced γ-ray emitting radioactive isotopes are[1] Na^{24}, Al^{28} and Mn^{56}.

The radioactive products that appear in fallout are the fission products and whatever radioisotopes are induced by weapon neutrons in materials in the immediate environment of the explosion. One such neutron-induced radioactivity occurs quite commonly because of the presence of U^{238} in nuclear devices.[4] The $U^{238}(n, \gamma)U^{239}$ reaction produces radioactive U^{239} which decays to Np^{239} by β-emission with a half-life of 23.5 min.[5] Because of its relatively short half-life, the radiations of U^{239} are of importance only in the very early times after detonation and, in many cases, are not of significant proportions by the time the fallout

has reached the ground. In addition, U^{239} has only a low energy 73 keV photon radiation which is of little significance when compared to the higher-energy γ-radiations emanating from the fission products. However, radioactive Np^{239}, the next isotope in the chain, decays with a half-life of 2.3 days, and has relatively prominent photon groups at 106, 228 and 278 keV; so it is possible for the radiations from Np^{239} to appear in significant quantities in fallout.

The magnitude of the dose received by an individual in the fallout radiation field depends on the number of quanta emitted by the source radioactive nuclei, the energy of these quanta and the degree to which material in the vicinity of the source absorbs or scatters the radiations.[6] The effectiveness of shielding material such as the walls of buildings is influenced by the first two factors. In any passive defense situation neither of these factors is under the control of either the civilian or the military population that must contend with the radiation. Thus exact predictions of the relative distribution of fallout or of the energy spectral characteristics of the radiation are impossible. Several models for predicting fallout distribution have been developed,[7–8] and some have been found to give reasonable agreement with experimental observation. The only currently generally available predictions of related γ-ray energy spectra are calculations[9–11] of the radiations from slow neutron fission of U^{235}. Several experimental measurements have been made of the spectral characteristics of the γ-radiation from fallout.[12–16] It is the purpose of this paper to give a general summary of these measurements, to discuss the confidence limits that can be placed on the experimental measurements, and to compare experimental measurements with calculated predictions.

EXPERIMENTAL MEASUREMENTS

The samples observed in the experimental studies were collected in a variety of types of fallout collectors or as cloud samples at the various weapons tests from 1954 through 1957. Within the ability of each collecting device to get a representative sampling of the radioactive debris, all samples are representative of the actual fallout that came down in the vicinity of the detonation.

For measurement of γ-ray spectra, samples were prepared either as unsealed powders, as a residue from the evaporation of liquids in which samples had been dissolved, or in other appropriate forms. Radioactive sources to be measured were placed in the bottom of a test tube and located above a $\frac{1}{2}$-in. diameter lead collimator, as illustrated in Fig. 1.

The γ-ray spectra from the radioactive samples were measured[12–16] by a scintillation spectrometer, using a 4 in. diameter by 4 in. long cylindrically shaped NaI(Tl) crystal, and a DuMont type 6364, 5 in. diameter photomultiplier tube in the geometrical arrangement illustrated in Fig. 1. The pulse-height distribution from this system was analyzed by: (a) a single-channel analyzer, the information from which was recorded on a strip chart recorder (1954–1955–1956); (b) a twenty-channel analyzer, which could be operated to record five sets of adjacent intervals of the spectrum to give a total of 100 channels of information which was recorded in digital form on registers (1955); or (c) a hundred-channel analyzer, the data from which was recorded on a printer readout in digital form (1957).

The information which formed the output of the analyzer was a series of full-energy peaks and their associated continuous Compton distributions. Since there are many γ-rays producing the fallout spectral distribution, the lower-energy peaks and Compton distributions appear superposed on the higher-energy distributions, as illustrated in Fig. 2 for a selection of three γ-rays of different energies. To determine the relative intensities of the γ-rays in the fallout spectral distribution either a sort of unfolding must be done, in which individual pulse-height distributions are successively peeled away from the spectrum,[17] or some other system of analysis[18] must be used.

RESULTS

The experimental results reported are accumulations of the results from the 1955, 1956 and 1957 measurements. Since many of the measurement techniques had not been perfected at the time of the 1954 measurements, the reliability of those measurements must be questioned and therefore are not included in the results. No selection was made with regard to yield of weapons for the results reported here. The information has been gathered from radioactive samples from weapons having a range of yields from kilotons to several megatons.

In all the experimental spectra, γ-rays from the fission products and from Np^{239} are observed. On occasion some other neutron-induced activity appears, but, because of its

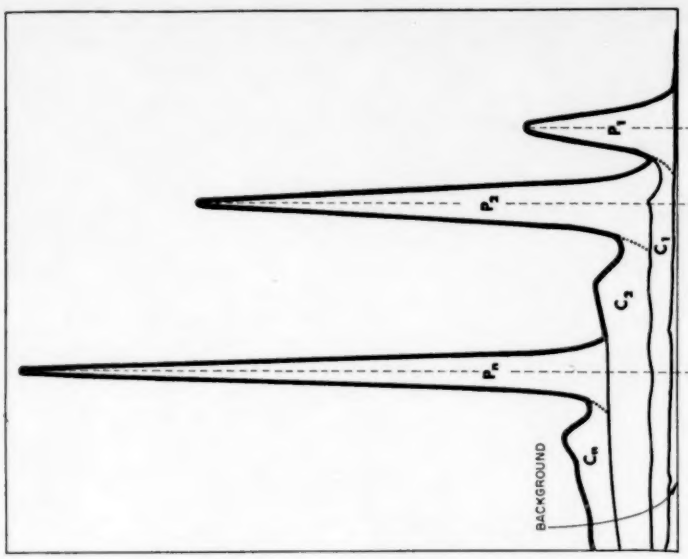

FIG. 2. Typical pulse-height distributions from three γ-rays, showing full-energy peak and Compton electron distribution for each γ-ray, and manner in which lower-energy γ-ray distributions are superposed on Compton distributions of higher-energy γ-rays.

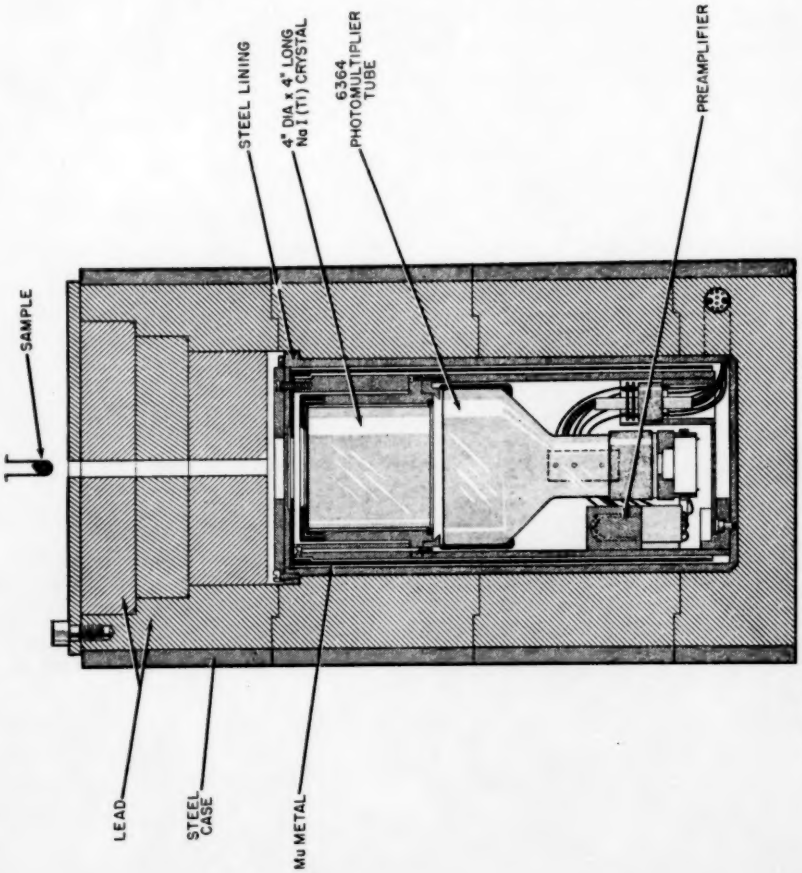

FIG. 1. Cross-sectional view of lead collimator and NaI(Tl) crystal-photomultiplier system, showing general location of radioactive fallout samples at time of measurement.

irregular appearance, any γ-radiation that can be positively identified as originating from neutron-induced activity other than Np^{239} has been eliminated from the data. Such neutron-induced activity was observed in only a very few cases so the elimination of these γ-rays from the data appears justified.

Experimental information on the more intense radiations is plotted for individual samples as circles in Figs. 3 and 4 in the form of percentage of total number of photons as a function of time. The radiations plotted in Figs. 3 and 4 account for between 65 and 85 per cent of the total number of photons in each sample measurement. Other γ-radiations which appear in the spectrum at some time after 6 hr following detonation are radiations at about 20–30 keV (1–10 per cent of the photons), about 60 keV (1–10 per cent), 140 keV (3–8 per cent), 330 keV (1–6 per cent), 370 keV (1–6 per cent), 450 keV (1–6 per cent), 578–610 keV (0–4 per cent), 1030 keV (0–3 per cent), and 1380 keV (0–10 per cent). Most of the 20–30 keV radiations are believed to be characteristic X-rays[19] formed as a result of internal conversion in the radioactive decay although a few very low-energy γ-rays may be present. The radiation at 1380 keV has a half-life of about 3 hr, appears only in the early times[16] and is believed to be Sr^{92}. A number of photons of energy higher than 1.6 MeV appear within the first 6 hr after detonation. For example, at 2 hr there appear to be approximately ten γ-rays distributed between 1.6 and 3 MeV, amounting to approximately 17 per cent of the total photons. At 3 hr this distribution of γ-rays appears to amount to approximately 15 per cent of the total number of photons.

COMPARISONS WITH CALCULATIONS

There have been three relatively recent calculations of the expected spectra of γ-ray energies as a function of time after slow neutron fission of U^{235}. PERKINS and KING[9] divide the fission-product γ-ray spectra into the same energy groupings as MOTEFF[20], and consider not only the γ-ray energy release due to instantaneous thermal neutron fission of U^{235} but also the γ-rays expected from the fission products resulting from reactor operating times of 1, 10, 100 and 1000 hr. The calculations by NELMS and COOPER[10] and by DOLAN[11] were made for instantaneous fission only. Their results are tabulated in smaller energy intervals than the results of PERKINS and KING. For these reasons the calculations by NELMS and COOPER and by DOLAN are used here for comparison purposes. Supposedly, NELMS and COOPER's results and DOLAN's results should be the same.

Since calculations have been made for fission product activity only, the γ-radiation from Np^{239} does not appear in any calculation of expected radiations. The 100 keV, 225 keV and 275 keV radiations in the experimental spectra are at least partially the result of the decay of Np^{239}, so, to eliminate the effect of Np^{239} radiations, comparisons between experiment and calculation have been made using only the energy intervals above 286 keV. These results are plotted in Figs. 5–12.

Both NELMS and COOPER, and DOLAN have made two types of tabulations of the γ-radiation from the U^{235} fission products as a function of time. In one tabulation the radiations from all radioactive fission-product nuclides are included, but, in the other, radiations from selected volatile fission products have been removed, the thought being that these volatile materials are apt to be lost in any collection of fission products that are exposed to the air, such as, for example, fallout. NELMS and COOPER remove the radiations from the bromine, krypton, ruthenium, iodine and xenon isotopes, whereas DOLAN removes the photons from krypton, xenon and iodine and those fractions of the daughter products that result from the decay of these elements. In those cases in which a daughter product has an independent fission yield, that fraction that is formed independently is retained in DOLAN's calculation. It is not completely clear whether these have been retained or removed from the calculations of NELMS and COOPER. In Figs. 5–12 calculational results, using the yield theory of GLENDENIN *et al.*[21], are shown both for the complete radiation spectrum, and for the spectra that result after the volatile elements have been removed. Calculations were made at specific time intervals following instantaneous fission. It is assumed that the results are continuous

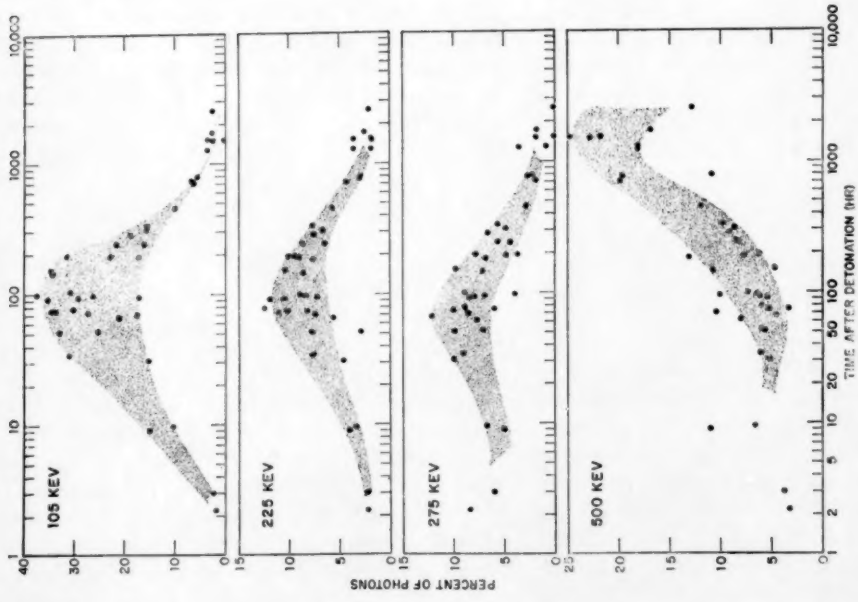

FIG. 3. Percentage of photons in total γ-ray spectrum having energies of 105 keV, 225 keV, 275 keV and 500 keV. Solid circles are experimentally measured results from individual fallout samples. Shaded areas have been used to illustrate the range of photon abundances at any given time after detonation.

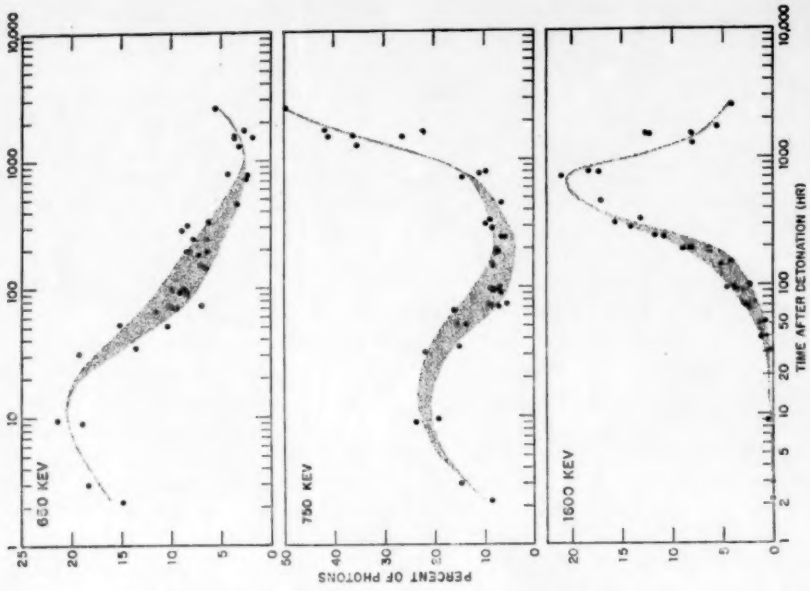

FIG. 4. Same as Fig. 3 except for photon energies of 650 keV, 750 keV and 1600 keV.

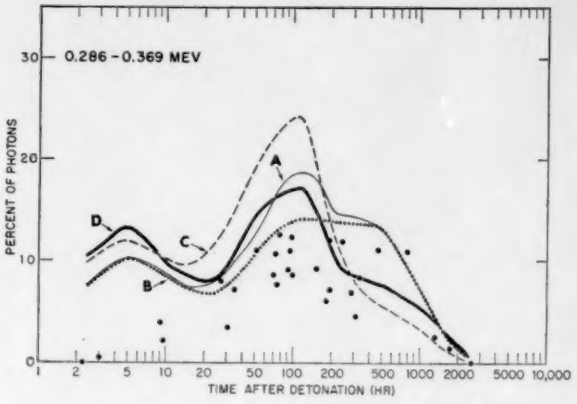

FIG. 5. Percentage of photons in spectrum above 0.286 MeV having energies in the interval between 0.286 MeV and 0.369 MeV, plotted as a function of time after detonation that spectral determinations are made. Solid circles are experimentally measured results from individual fallout samples. Curve *A* is based on calculations by DOLAN[11] of γ-radiation expected from products of slow neutron fissioning of U^{235}. Curve *B* has been derived from the same type of calculations made by NELMS and COOPER[10]. Curve *C* shows results of calculations by DOLAN after removal of radiations from iodine, krypton and xenon isotopes and a fraction of the radiations from their daughter products. Curve *D* is based on the calculations by NELMS and COOPER for same fission products after removal of radiations from bromine, krypton, ruthenium, iodine and xenon isotopes.

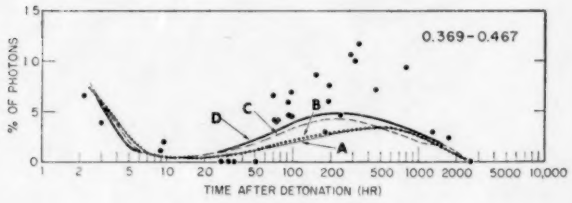

FIG. 6. Same as Fig. 5, except for energy interval between 0.369 MeV and 0.467 MeV.

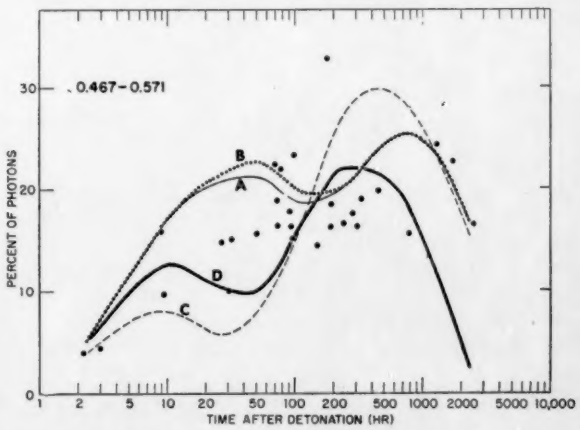

FIG. 7. Same as Fig. 5, except for energy interval between 0.467 MeV and 0.571 MeV.

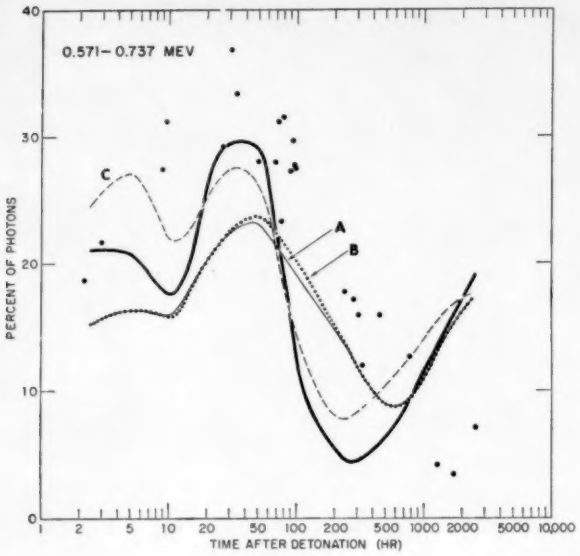

FIG. 8. Same as Fig. 5 except for energy interval between 0.571 MeV and 0.737 MeV.

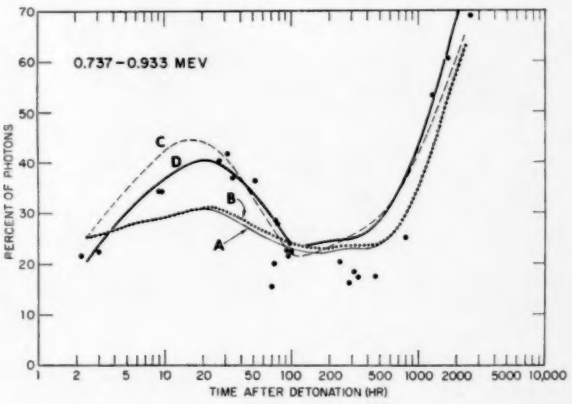

FIG. 9. Same as Fig. 5 except for energy interval between 0.737 MeV and 0.933 MeV.

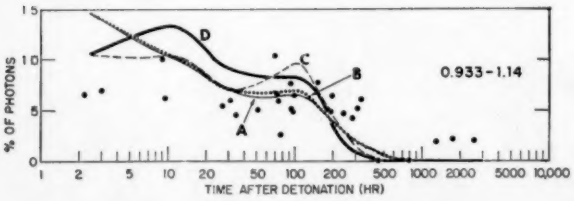

FIG. 10. Same as Fig. 5 except for energy interval between 0.933 MeV and 1.14 MeV.

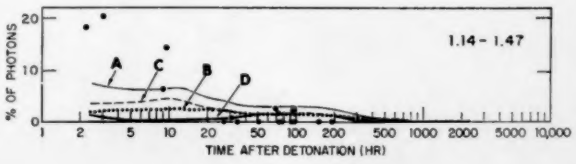

FIG. 11. Same as Fig. 5 except for energy interval between 1.14 MeV and 1.47 MeV.

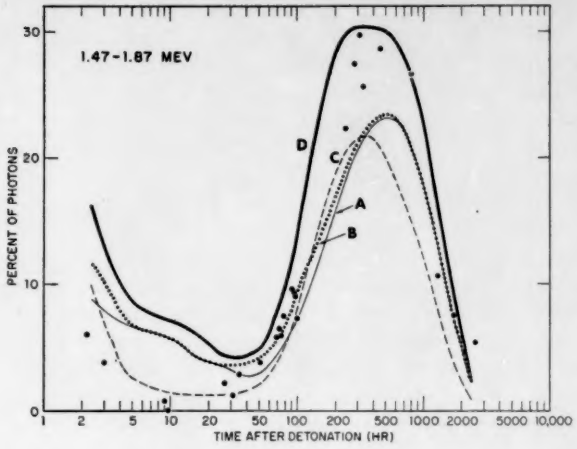

FIG. 12. Same as Fig. 5 except for energy interval between 1.47 MeV and 1.87 MeV.

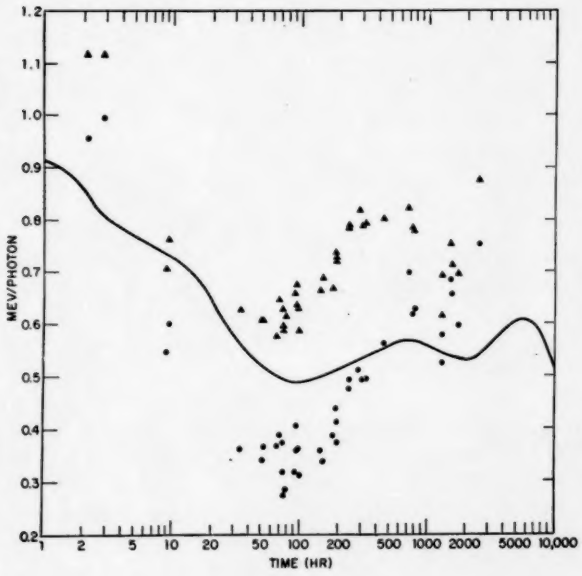

FIG. 13. Mean energy per photon as a function of time after detonation. Solid line is calculation by MILLER[22] for fission-product γ-radiation. Circles are experimentally determined values from individual fallout sample γ-radiation. Triangles are experimentally determined values for radiations above 0.286 MeV, as explained in the text.

between points, but since the exact shape is not known, lines in these figures have been rather arbitrarily drawn to pass through each of the calculational results. The experimental results appear to follow the same general shape as the calculated expectations, but the agreement is often not really good.

MILLER[22] has calculated the expected mean energy per photon for fission-product γ-radiation. The continuous line in Fig. 13 gives the results of his calculations. The experimental points for the spectra including Np^{239} are shown as closed circles. Since the experimental data include all radiations it is expected that the low-energy Np^{239} radiations will cause the mean energy per photon to fall below the calculated values. This effect is observed at the intermediate times on the plot. If the Np^{239} radiations at 105 keV, 225 keV and 280 keV are removed from the spectra, and the 20–30 keV and 60 keV radiations are also removed, based on the assumption that they are partially

contributed by Np^{239}, and furthermore can be readily shielded, the triangles in Fig. 13 are obtained. The mean energy is probably too high at 2.2 and 3.0 hr for it appears that the radiation at 280 keV is more intense than would be expected strictly from the Np^{239} contribution. Later time contributions (after 10 hr) appear to follow the general shape of the curve calculated by MILLER, but at a slightly higher energy and seem to fluctuate more closely around a mean energy of 0.7 MeV. The mean γ-ray energy is used here as a simple means for comparing calculations with experimental observations. Under no circumstances should the quotation of a mean γ-ray energy be misconstrued to be a single energy that can be used in shielding calculations or other physical processes for which the actual distribution of spectral components must be considered.

DISCUSSION

In this summary γ-ray spectra from individual fallout samples experimentally measured at different times after weapon detonation have been tabulated in selected energy intervals and plotted in terms of percentage of photons as a function of time. In the results general trends can be observed in the γ-ray spectra which do not appear to be a function of weapon yield. There is considerable spread in the relative photon yield for each energy at least at some times. Some of this spread appears to depend on the amount of neutron-induced Np^{239} activity, and hence the amount of 100 keV, 225 keV and 280 keV radiations, that exist in the spectrum. If these radiations are eliminated, and only the fission products considered, the spread may not be as great as in the original cumulation of data, but, even in this case, some variation appears to be inherent in the spectral characteristics of the radiations.

The equipment used for these measurements has also been used to measure the relative intensities of γ-radiations from selected individual radioactive isotopes. In the study of these isotopes the measurement errors have been shown to be no greater than a few per cent. On this basis, assignment of relative measurement errors of no more than 10 per cent[12–16] for the more intense γ-rays seems reasonable. The less intense γ-rays must be assigned appropriately larger relative errors depending on the accuracy with which the area of full-energy peaks in the pulse-height spectra can be measured.

Even after the removal of the Np^{239} radiations, the deviation of individual experimental results from any appropriately selected mean value is usually larger than is expected from any known source of possible error. This deviation is usually of about the same magnitude as the differences between experimental results and calculated predictions. As stated previously, it is encouraging that the general shape of both experimental and calculational distributions as a function of time for most energy intervals are similar. It would appear, therefore, that a reasonably close estimate of the γ-ray spectral characteristics of fallout can be made by an appropriate mixture of the fission-product radiations and the radiations from Np^{239}.

It appears that some variation in the spectral characteristics of the fallout γ-radiation results from the effects of fractionation of the radioactive products at very early times after detonation. This process is not sufficiently well understood to make reliable predictions of its effect on the distribution of radioactivity in fallout. However, there is sufficient knowledge about the fractionation process to realize that the relatively simple assumptions made by NELMS and COOPER, and by DOLAN are probably inadequate. The amount of volatile or gaseous material that is separated from the fallout material appears to depend, for instance, on the half-life of the material. Longer half-life materials have a greater percentage of removal than shorter-lived isotopes. There may be a number of factors in the chemical properties of the materials that enhances the fractionation process. Because of the large number of possible variables in any single detonation, it is not expected that, even if only fission products were present in fallout, the γ-ray spectral characteristics would remain absolutely the same from sample to sample. It may even be considered surprising that trends in the experimental data as a function of time are as clear cut as they are for such a variety of weapon types and yields.

The results presented in this paper should be considered only as a general type of expected

γ-ray spectral distribution. Individual nuclear weapons may, because of construction designs and the nature of the detonation environment, especially with regard to its influence on neutron-induced radioactivity, produce fallout spectra of quite different characteristics. The exact nature of the γ-ray spectral characteristics cannot be given ahead of time, especially in a real wartime situation. It is hoped, though, that the results presented here indicate at least the general γ-ray energy spectral characteristics that might be expected in fallout from a nuclear weapon, and the degree of divergence between calculated and observed spectral characteristics.

Acknowledgements—It was possible to gather the experimental data from which this summary report has been derived only because of the combined efforts of a large number of personnel at USNRDL. Those primarily responsible for the successful accumulation and analysis of the data were R. L. MATHER, W. E. THOMPSON, F. M. TOMNOVEC, J. M. FERGUSON, L. A. WEBB, R. F. JOHNSON, P. R. HOWLAND, R. A. TAYLOR, W. BUSER and R. EHAT.

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COBALT-60 BOMBS

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Abstract—Based on stated assumptions, the ratio of potential external energy releases from fission products in "local" fallout compared to the cobalt-60 that may be produced can vary from a factor of 40, based on an exposure period from 1 hr to 1 month after a detonation, to 0.17 based on an exposure period from the twenty-fourth hour to 35 years. It is concluded that other factors, i.e. fission yields and position of firing, would be more critical in determining the external γ-hazard than the added cobalt-60.

One megaton (total yield) detonated in the air might create enough cobalt-60 to produce a theoretical total dose of 0.17 r in the most heavily contaminated band of the earth (30°–60° north latitude), with weathering and shielding effects reducing this by as much as a factor of 10. Decontamination measures could reduce this dose further. Other doses may be calculated as a linear function of the tonnage assumed. Thus, the estimated doses could be significant in long-term effects but not lethal in the sense of wiping out a worldwide population. Mathematically, one can go to ever higher and higher assumed tonnage and cobalt-60 production until a lethal range is reached.

THE subject of cobalt-60 bombs in nuclear warfare has been raised in the past. The following brief discussion attempts to place this concept in some perspective.

SUMMARY STATEMENTS

I. "*Local*" *fallout*

The total theoretical potential external energy releases from cobalt-60 may exceed those from fission products only when calculations are made for periods greater than 1 year (based on the assumptions used in the calculations below).

Table 1. *Ratio of total energy releases: fission products/cobalt*-60

	Start of time period (hr)	
	1	24
End of time $\frac{1}{12}$	40.0	11.0
period (years) 1	4.0	1.3
35	0.5	0.17

The dose *rates* from the fission products greatly exceed those of the Co^{60} at early times after a detonation (factor of about 5900 $\pm$ 40% at $H + 1$), creating a potentially more hazardous situation in terms of lethality or radiation sickness. Also, with the relatively long time for delivery of a given dose from Co^{60}, decontamination measures could be instituted to reduce this potential exposure. Further, fission products (such as Sr^{90} and Cs^{137}) would be more significant than Co^{60} as internal emitters, since the latter has such a low percentage uptake into the body.

The "advantages" of using Co^{60} are not significant enough to warrant the effort, i.e. large nuclear bombs, with high percentages of their energy derived from fission, fired on the surface would create local fallout patterns of such relatively high intensities that the added effect of Co^{60} would not be the critical factor.

II. "*Worldwide*" *fallout*

Data from past nuclear tests strongly indicate that detonations in the general latitude of United States and Europe would result in the fallout being distributed roughly in a Gaussian

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distribution, with the mean at about 45° north latitude and a standard deviation of 10°–15°, i.e. about 75 per cent of all the "worldwide" fallout would occur between 30°–60° north latitude and would thus represent the most heavily contaminated areas of the world. To achieve maximum fallout of Co^{60} "worldwide" (30°–60° north latitude) the nuclear detonations would have to be air bursts. Under these conditions, the potential external γ-exposure from the Co^{60} would greatly exceed that from fission products.

Without regard to decontamination measures, or weathering and shielding factors, a 1 Mton detonation in the air might create enough Co^{60} to produce a theoretical dose *rate* on the ground of about 0.019 r/year at the peak point (3.2 years after the detonation), with a total energy release of about 0.19 r in 35 years in the 30°–60° north latitude zone. Weathering and shielding factors might reduce the above values by a factor as great as 10.[1] Further reduction by contamination measures would vary greatly depending upon the efforts employed. Because of the relatively long half-life of Co^{60} (in the sense used here), extended periods of time would be available in which to perform decontamination measures before an appreciable fraction of the potential 35-year dose were delivered.

Other dose rates and doses may be calculated as a linear function of the tonnage assumed. Thus, the estimated actual cobalt-60 doses could be significant in such terms as increase rate of leukemia and bone tumor, life shortening and in genetic considerations, but not lethal in the sense of wiping out a worldwide population. Of course, one can mathematically go to ever higher and higher assumed tonnage for the detonations and production of cobalt-60 until a lethal range is reached.

CALCULATIONS

I. "*Local*" *Fallout*

A. *Comparison of total energy releases*

Fission products vs. cobalt-60:

(1) Assume $\frac{1}{2}$ Mton fission and $\frac{1}{2}$ Mton fusion, then

$\frac{1}{2}$ Mton (fission) $\rightarrow 7.2 \times 10^{25}$ fissions

(2) At 1.0 hr after detonation, one fission would result in fission products releasing 0.51 MeV/hr, or a total of 1.7 MeV from 1 hr to 35 years.[2]

$(7.2 \times 10^{25})(0.51) \rightarrow 3.7 \times 10^{25}$ MeV/hr per 1 Mton total yield

$(7.2 \times 10^{25})(1.7) \rightarrow 1.2 \times 10^{26}$ MeV (1 hr to 35 years) per 1 Mton total yield

(3) Similar calculations at 24 hr (for example), $(7.2 \times 10^{25})(0.58) \rightarrow 4.2 \times 10^{25}$ MeV (total energy from fission products twenty-fourth hour to 35 years) per 1 Mton total yield

(4) Assume

1 Mton (total yield) $\rightarrow 2 \times 10^{26}$ neutrons escape
$\rightarrow 1 \times 10^{26}$ atoms Co^{60} produced

Since each atom of $Co^{60} \rightarrow 2.5$ MeV
then $(1 \times 10^{26})(2.5) \rightarrow 2.5 \times 10^{26}$ total MeV energy release from Co^{60} atoms, per 1 Mton total yield.

B. *Comparison of dose rates*

Assume

1 kton (total) $\rightarrow 2 \times 10^{23}$ neutrons escape
$\rightarrow 1 \times 10^{23}$ atoms Co^{60} produced
$\rightarrow \frac{1}{2}$ kton fission (7.2×10^{22} fissions)

(1) Based on fission products from 1.0 kton (1 kton total, half fission, half fusion) placed on a square mile:

1000 r/hr $\pm$ 40 per cent at $H + 1$[4]

(2) 1 kton (total) $\rightarrow 1 \times 10^{23}$ atoms Co^{60} produced (assumed)
$\rightarrow 1.15 \times 10^{4}$ c Co^{60}

1.15×10^{4} c $Co^{60}/mile^2 \rightarrow 0.17$ r/hr (infinitely flat plane)

(3) $\frac{1000}{0.17} \rightarrow 5900 \pm 40$ per cent greater γ-dose *rate* from fission products than Co^{60} at $H \pm 1$

or $\rightarrow$ roughly 2 ± 40 per cent at $D + 30$

II. "*Worldwide*" *Fallout*

Assume

1 Mton $\rightarrow 2 \times 10^{26}$ neutrons

1×10^{26} neutrons are captured by Co^{59}

(1) 1×10^{26} neutrons $\rightarrow 1 \times 10^{26}$ atoms Co^{60} $\rightarrow 1.15 \times 10^{7}$ c

(2) $(1.15 \times 10^{7})(0.05) \rightarrow 5.8 \times 10^{5}$ c to troposphere (assumed)

$(1.15 \times 10^{7})(0.95) \rightarrow 1.1 \times 10^{7}$ c to stratosphere (assumed)

(3) $\dfrac{(5.8 \times 10^{5})(0.75)}{36 \times 10^{6}} \rightarrow 1.2 \times 10^{-2}$ c/mile^2 deposited 30°–60° north latitude from troposphere (within several weeks)

(4) If all the Co^{60} remained in stratosphere, then total energy expended:

$$\frac{A_0}{\lambda_2} \rightarrow 7.6A_0$$

where: A_0 = original activity;
λ_2 = radiological decay constant.

With an assumed average time of residence of 2 years*, actual energy expended on the ground:

$$\left[\frac{\lambda_1}{\lambda_1 + \lambda_2}\right] 7.6A_0 = 6.0A_0,$$

or about 79 per cent of theoretical total where λ_1 = residence time constant.

(5) $\dfrac{(1.1 \times 10^{7})(0.79)(0.75)}{36 \times 10^{6}} \rightarrow 1.8 \times 10^{-1}$ c (equivalent)/mile^2 deposited 30°–60° north latitude from stratosphere

(6) $(1.2 \times 10^{-2}) + (1.8 \times 10^{-1}) \rightarrow 1.9 \times 10^{-1}$ c (equivalent)/mile^2 total

(7) 10^{6} c/mile^2 of $Co^{60} \rightarrow 15$ r/hr (3 ft above infinitely flat plane)

$\dfrac{(1.9 \times 10^{-1})(15)}{10^{6}} \rightarrow 2.8 \times 10^{-6}$ r/hr (equivalent)

$\rightarrow 2.5 \times 10^{-2}$ r/year (equivalent)

(8) $D = RT_m$

$D = (2.5 \times 10^{-2})(7.6)$

$= 0.19$ r (in about 35 years)

where: D = total theoretical dose;
R = dose rate;
T_m = mean life.

(9) $\dfrac{0.19}{10\dagger} \rightarrow 0.019$ r estimated dose

(10) To determine time when there will be a maximum Co^{60} on ground from stratospheric fallout and the amount:

$$\frac{dq}{dt} = \lambda_1 Q_0 \exp\left[-(\lambda_1 + \lambda_2)t\right] - \lambda_2 q$$

where: q = amount on earth at time t;
Q_0 = original amount in stratosphere;
λ_1 = constant of storage time in stratosphere;
λ_2 = radiological decay constant.

$$q = Q_0 \exp(-\lambda_2 t)[1 - \exp(-\lambda_1 t)]^2$$

Differentiating and setting equal to zero:

$$\exp(-\lambda_1 t) = \frac{\lambda_2}{\lambda_1 + \lambda_2}$$

$t = 3.24$ years

Substituting

$$q = 5.8 \times 10^{6} \text{ c/Mton}$$

from stratosphere, present on the ground at maximum (75 per cent of which is in the 30°–60° north latitude or 0.12 c/mile^2);

plus

$(1.2 \times 10^{-2})[\exp(-0.131 \times 3.24)] \rightarrow 7.8 \times 10^{-3}$ c/mile^2 (30°–60°N)

from tropospheric fallout at 3.2 years;

$(1.2 \times 10^{-1}) + (7.8 \times 10^{-3}) \rightarrow 1.3 \times 10^{-1}$ c/mile^2 (30°–60° N)

at 3.2 years;

$\rightarrow 1.9 \times 10^{-2}$ r/year *rate* at 3.2 years.

* The residence time may vary[3] but, as an upper limit, the potential energy expenditure on the ground, of course, could not exceed 100 per cent, i.e. an increase of 21 per cent over these calculations.

† Assumed weathering and shielding factor.[1]

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A CORRELATION ANALYSIS OF THE RELATIONSHIP BETWEEN WEATHER VARIABLES AND ATMOSPHERIC RADIOACTIVITY*

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Abstract—Correlation analysis between various surface weather parameters and samples of atmospherically borne radioactive material has been performed. The material collected was primarily of natural origin (i.e. radon and thoron). Statistically significant correlations exist between wind speed at the time of sampling and between temperature, dew point, visibility and the north–south component of the wind vector prior to the time of sampling. The analysis suggests that the natural radioactive material is transported into the Austin area within the turbulence associated with a cold frontal passage and concentrated in the lower air layers by the ensuing elevated temperature inversion.

With the advent of large-scale nuclear bomb testing, increased attention has been focused on radioactive atmospheric pollutants and their associated biological hazards. Recently, emphasis has been placed on the study of nuclear detonation fallout, whereas it has long been established that significant amounts of radon (Rn^{222}) and thoron (Rn^{220}) in equilibrium with their daughter products are found existing naturally in the atmosphere. These gaseous elements are members of the uranium series and thorium series of decay chains, respectively; therefore, most of the naturally occurring radioactive material in the atmosphere represents the decay of uranium and thorium in the earth's crust which has diffused as radon and thoron into the atmosphere. Their presence in the atmosphere has been measured at numerous widely scattered locations.[1–5]

Radon in equilibrium with its daughter products exhibits an apparent half-life of approximately 30 min. While the primary mode of decay in the uranium and thorium series is by α-emission, atmospheric radon in equilibrium exhibits an activity that is primarily β-emission from Ra *B* and Ra *C*. Thoron in equilibrium with its daughter products shows a longer apparent half-life than the radon series or one nearly 10.5 hr. The radiation of this series is primarily the beta of Th *B* and Th *C*.[8]

Interest in atmospherically borne radioactivity at this laboratory originated from health physics considerations. Air sampling in the Mixed Irradiation Facility of this laboratory was conducted daily in order to ascertain that none of the PoBe sources in use had ruptured. A similar air sample was taken in open air to provide a concurrent radiation baseline. Each sample was evaluated for radioactivity. Extreme variability of the latter sample was noted. Decay curves for the samples usually indicated an apparent half-life of 30 min. Assuming radon in equilibrium with its daughter products to be the primary active element, then the possibility that the variation in the outdoor samples might be related to weather phenomena led to the present study.

Statistical methods were applied to the count rates obtained from the outdoor air samples in order to ascertain, if possible, any relationships between the level of radioactivity and certain weather parameters. Correlation coefficients

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between atmospheric radioactivity as determined from air samples and the weather parameters were calculated. From such analysis it was hoped to obtain a pattern of climatological factors, if any exists, affecting the variation observed and to locate the source of the contributing material, if a unique one exists. The latter goal was prompted by the presence of high concentrations of uranium and thorium located in a region north-east of Austin known as the Llano Uplift.[6]

MATERIALS AND METHODS

The samples of the atmospherically borne radiation contamination were taken by drawing air through a Whatman 41 H filter with an ORNL designed air sampler pumping air at a flow rate of 22.2 l/min.[7] The samples were taken each work day morning at 0730 Austin Time (1330Z) for a $\frac{1}{2}$ hr sampling period. The filter paper samples were counted immediately for their radioactivity in a windowless gas-flow counter. An amplifier was inserted behind the counter so that by proper adjustment of chamber potential, gain and pulse-height discrimination, the counter could be operated as a Geiger–Muller counter or a proportional counter. When operated as a proportional counter, the pulse-height discrimination was biased to count the α-disintegrations within the sample; of course, as a Geiger–Muller counter, all ionizing radiations were counted. Radiation levels for these samples were recorded in counts per minute. The apparent half-life of the high count rate material normally was approximately 30 min. Occasionally the apparent half-life was 2 or 3 hr. If only radon and thoron together with their decay products existed in the atmosphere, then the variation in apparent half-life could be used to determine the ratio of the total activity present from each decay scheme. However, since the longer apparent half-life samples seemed most prevalent during nuclear test periods, the possibility of nuclear fallout restricted the use of these data to identifying the presence of the radon series.

The weather data used were obtained from U.S. Weather Bureau code Sm 32 teletype reports listing data taken at 0700Z (Greenwich Mean Time) and 1230Z for the Austin area. In addition, weather variables were taken for Waco, approximately 100 miles north, and for San Antonio, approximately 80 miles south of Austin.

RESULTS

In a preliminary analysis, correlation coefficients for atmospheric Geiger–Muller count rates and several Austin weather parameters were found. Weather factors tested were wind speed, dew point, cloud cover, temperature, visibility, north–south component of the wind speed, east–west component of the wind speed, pressure gradient between Waco and San Antonio, and the change in pressure during the preceding 24 hr. Correlation coefficients were calculated for the weather variables at 0700Z; 1230Z; 1230Z, 24 hr earlier; 1230Z, 48 hr earlier; 1230Z, 72 hr earlier. Weather data tested in the analysis were all ground-level observations. Table 1 presents a summary of

Table 1. *Correlation coefficients for G.–M. count rates vs. Austin weather parameters for* 1955

Weather factors	1230Z	0700Z	1230Z (−24 hr)	1230Z (−48 hr)	1230Z (−72 hr)
Wind speed	n.s.	−0.247	n.s.	n.s.	n.s.
Dew point	n.s.	n.s.	n.s.	−0.203	n.s.
Cloud cover	−0.248	n.s.	n.s.	n.s.	n.s.
Temperature	n.s.	n.s.	−0.151	n.s.	n.s.
Visibility	−0.278	−0.174	n.s.	n.s.	n.s.
North–south component of wind	n.s.	n.s.	n.s.	n.s.	n.s.
East–west component of wind	n.s.	n.s.	n.s.	−0.165	n.s.
Pressure gradient between Waco and San Antonio	n.s.	n.s.	n.s.	n.s.	n.s.
Change in pressure during preceding 24 hr	n.s.	n.s.	n.s.	n.s.	n.s.

n.s. = not significant at 0.05 level of significance.

the correlation coefficients found in this preliminary analysis.

Similar correlation coefficients were computed for proportional count rates vs. the same weather parameters and are listed in Table 2.

After completing the above, a similar study was made for the data gathered during 1956. The results were essentially the same as those found for the 1955 data.

With the completion of the computation of the 1955 and 1956 data, it was decided to test only those variables for the 1957 and 1958 data that were statistically significant in the original study. Further, data were exempt from the new work when taken during the two nuclear test periods in Nevada on days wherein significantly high dose rates were found (G.–M. count rate >300 c/min). The weather information used was:

wind speed at 0700Z;
wind speed at 1230Z, 48 hr earlier;
cloud cover at 0700Z;
cloud cover at 1230Z, 48 hr earlier;
temperature at 1230Z, 24 hr earlier;
temperature at 1230Z, 48 hr earlier;
visibility at 1230Z;
visibility at 1230Z, 48 hr earlier;
the north–south component of the wind speed at 1230Z, 48 hr earlier;
the east–west component of the wind speed at 1230Z, 48 hr earlier;
dew point at 1230Z, 24 hr earlier;
dew point at 1230Z, 48 hr earlier.

The resulting correlation coefficients are presented in Tables 3–6.

Multiple correlation coefficients were obtained for the two types of radiation measurements. These values are listed in Table 7.

A generally accepted convention, in order for a regression equation to be useful, is that it should exhibit a multiple R on the order of 0.9 or greater. The results above do not satisfy this criterion; however, prediction equations were developed for the proportional count data. This enabled the significance of each variable to be tested as a predictor by eliminating that variable from a recalculation of the multiple R correlation coefficient. This was done and those variables which did not significantly affect the multiple R were eliminated. For 1957, the useful predictors were reduced to three variables. The resulting equation was

$$Y = 36.59 - 0.8973f - 0.5004t - 0.5932V \quad (1)$$

where Y is the proportion count rate; f is the wind speed at 0700Z; t is the temperature at 1230Z, minus 24 hr; V is the visibility at 1230Z, minus 48 hr. For equation (1), the multiple R was calculated as 0.497.

For 1958, it was found that the multiple R for these same few predictors for proportional count data was 0.303. Wind speed alone at 0700Z gave essentially as good a prediction, however. The product moment r between the proportional count, 1958 and wind speed at 0700Z was −0.255, giving the prediction equation

$$Y = 23.27 - 1.0012f \quad (2)$$

where again Y is the proportional count rate and f is the wind speed at 0700Z.

Finally, it was of interest to determine how well the 1957 prediction equation could forecast the 1958 proportional count rates, using the observed values of the weather variables for 1958 yielding a correlation coefficient of 0.261. The reason for the equally good prediction is that the coefficient of the wind speed is of the same order of magnitude in equations (1) and (2).

To obtain a visual picture of the effectiveness of equation (1), the average of five time-adjacent observed values was plotted against the corresponding average of five time-adjacent predicted values for the proportional count rate of the year 1957 (Fig. 1). Values were missing on Saturdays and Sundays plus an occasional missing value in one of the predictors. In case of a missing value for one of the predictor variables, the entire observation vector was omitted. Thus, the total sample number in which all predictors were present was 175.

DISCUSSION

It is readily discernible in the data presented that the correlations between several of the weather factors and atmospheric radioactivity show considerable variability from year to year. However, it is equally evident that for a

Table 2. Correlation coefficients for proportional count rates vs. weather parameters for 1955

Weather factors	1230Z	0700Z	1230Z (−24 hr)	1230Z (−48 hr)	1230Z (−72 hr)
Wind speed	−0.222	−0.293	n.s.	n.s.	n.s.
Dew point	−0.156	n.s.	−0.268	−0.305	−0.173
Cloud cover	−0.217	−0.183	−0.203	−0.164	n.s.
Temperature	n.s.	−0.182	−0.290	−0.275	n.s.
Visibility	−0.160	−0.218	n.s.	n.s.	n.s.
North–south component of wind	n.s.	n.s.	n.s.	n.s.	n.s.
East–west component of wind	n.s.	n.s.	n.s.	n.s.	n.s.
Pressure gradient between Waco and San Antonio	n.s.	n.s.	n.s.	n.s.	n.s.
Change in pressure during preceding 24 hr	n.s.	n.s.	n.s.	n.s.	n.s.

n.s. = not significant at 0.05 level of significance.

Table 3. Correlation coefficients for G.–M. count rates vs. Austin weather parameters for 1957

Weather factors	0700Z	1230Z	1230Z (−24 hr)	1230Z (−48 hr)
Wind speed	−0.362	**	**	n.s.
Cloud cover	n.s.	**	**	n.s.
Temperature	**	**	−0.422	−0.390
Visibility	**	n.s.	**	n.s.
NS wind speed	**	**	**	+0.349
EW wind speed	**	**	**	n.s.
Dew point	**	**	−0.359	−0.352

n.s. = not significant at 0.05 level of significance.
** Not tested.

Table 4. Correlation coefficients for proportional count rates vs. weather parameters for 1957

Weather factors	0700Z	1230Z	1230Z (−24 hr)	1230Z (−48 hr)
Wind speed	−0.341	**	**	n.s.
Cloud cover	n.s.	**	**	n.s.
Temperature	**	**	−0.355	−0.294
Visibility	**	n.s.	**	−0.154
NS wind speed	**	**	**	−0.195
EW wind speed	**	**	**	n.s.
Dew point	**	**	−0.300	−0.235

n.s. = not significant at 0.05 level of significance.
** Not tested.

Table 5. Correlation coefficients for G.–M. count rates vs. Austin weather parameters for 1958

Weather factors	0700Z	1230Z	1230Z (−24 hr)	1230Z (−48 hr)
Wind speed	−0.328	**	**	n.s.
Cloud cover	n.s.	**	**	n.s.
Temperature	**	**	−0.310	−0.246
Visibility	**	n.s.	**	n.s.
NS wind speed	**	**	**	n.s.
EW wind speed	**	**	**	n.s.
Dew point	**	**	−0.320	n.s.

n.s. = not significant at 0.05 level of significance.
** Not tested.

Table 6. Correlation coefficients for proportional count rates vs. weather parameters for 1958

Weather factors	0700Z	1230Z	1230Z (−24 hr)	1230Z (−48 hr)
Wind speed	−0.255	**	**	n.s.
Cloud cover	n.s.	**	**	n.s.
Temperature	**	**	−0.161	n.s.
Visibility	**	n.s.	**	n.s.
NS wind speed	**	**	**	n.s.
EW wind speed	**	**	**	n.s.
Dew point	**	**	−0.164	n.s.

n.s. = not significant at 0.05 level of significance.
** Not tested.

number of these weather variables the pattern of correlation remains remarkably consistent. Thus, for example, one finds in the analysis of each of the 4 years a consistently high correlation between atmospherically borne radioactivity and the wind speed at 0700Z; the temperature at 1230Z, minus 24 hr.

While the residual variance is too great to permit extensive theorizing, an analysis of the results can yield some notions concerning a possible physical mode. The data separate into two groups: the weather parameters near the sampling time (i.e. wind speed and visibility) and the parameters taken prior to sampling (i.e. temperature, dew point and north–south component of wind minus 48 hr). These results suggest a mode of transport into the Austin area and an enveloping process to hold the particulate matter brought in. A typical day of high atmospheric radioactivity would be one preceded by turbulent winds generally from the north, blowing the activity into the Austin area. This would then be followed by lower temperatures, decreased winds and lower visibility. These later conditions are consistent with the existence of a temperature inversion stabilizing the surface air, while the entire process is typical of a cold frontal passage and the resulting elevated temperature inversion in this area.

Table 7. Multiple R correlation coefficient for Geiger–Muller and proportional count rate

Count rate and year		Multiple *R* correlation value
1957	(G.–M.)	0.56
1957	(prop)	0.51
1958	(G.–M.)	0.48
1958	(prop)	0.38

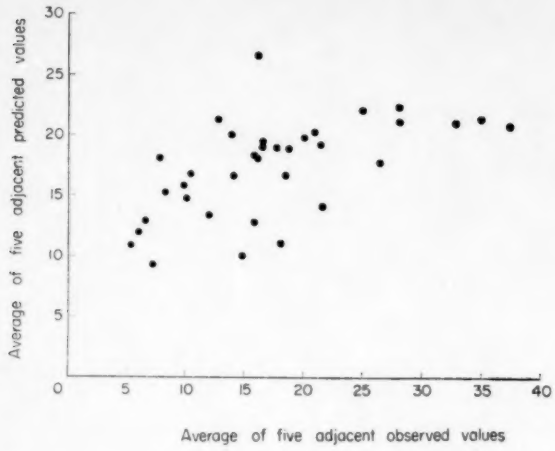

FIG. 1. A scatter diagram of the average of five time-adjacent observed values plotted against the t time-adjacent predicted values [equation (1)] for the 1957 proportional data.

The finding of LUCAS *et al.*[5] that a significant correlation exists between atmospheric radon concentrations and temperature inversions appears to corroborate much of the present study.

The prediction equations presented tend further to substantiate this view. Using an equation based on only part of the significant group of variables yielded almost as good a prediction as the formula incorporating the entire group of variables. Only those parameters necessary to describe a frontal passage through the Austin area appear to be important. That the predictions are not materially improved by the addition of the remainder of the individually considered significant variables implies that part of the residual variance is due to processes other than those involved in a local frontal passage.

The present study gives no indication concerning the source of the radioactivity measured. However, the general source of the atmospheric activity appears to be located at a considerable distance from Austin. This is indicated by the relationship of the north–south wind component minus 48 hr. This correlation may be accentuated because the prevailing southerly winds in the Austin area are from the Gulf of Mexico and therefore relatively free of radioactive material.[2] If the source were in the Llano Uplift, a significant correlation of the east–west component of the wind prior to sampling with count rate would be expected, since this area is approximately 50 miles north-west of Austin. In view of the relationships above, a specific area cannot be designated as the source of the radioactivity from the data.

The picture presented is entirely one of transporting and stabilizing the material in the Austin atmosphere. It is likely that many of the weather parameters at the source, if it exists as a specific locale, could affect atmospherically borne radioactivity. Since apparently this hypothetical source is distant from the Austin area or is extremely diverse, conclusions on "weather effects" in that area cannot be made from the present study.

A large amount of the residual or unexplained variation of the data in this study may be attributed to sampling techniques. The sampling procedure was originally designed for a different end point, with the result that the activities of the collected samples were quite low. Sampling a larger volume of air and optimizing the sampling time with respect to collecting rate and decay should materially lower the variance.

The only inference that can be drawn from this work is that a relationship exists between atmospheric radioactivity and certain weather parameters. The data neither give information on the physical processes involved, nor determine whether the relationship demonstrated is caused or whether atmospheric radioactivity and the weather parameters measured are each related to a still unmeasured variable.

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NOTES

Determination of Radioiodine in Milk after a Nuclear Accident*

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AFTER the Windscale accident the assay of milk for radiodine was carried out with scintillation counters.[1] In countries where the economy is largely agricultural, it might be difficult to find scintillation monitors in sufficient numbers in the event of extended contamination. Therefore, we suggest the use of gamma-counters of the Geiger–Müller type inserted into a pipe closed at the lower end, which may be introduced from the top into the full milk churn. A counter made of thick chromium steel, 13 cm long with a diameter of 4 cm, is very suitable. (The anode in longer tubes tends to be too fragile for continuous handling.)

Under normal circumstances the background of such a counter inside a churn filled with water amounts to about 130 counts/min, to which in the case of milk a few counts/min are added due to the potassium content. An amount of I^{131} in the milk corresponding to the maximum permissible concentration of 0.065 μc/l.[2] gives about 460 counts/min above background; this count rate is sufficient for accurate determination by a few minutes' counting.

If the counter is moved a few centimeters in a vertical or horizontal direction from the center of the churn the counting rate drops somewhat (amounting to a difference of somewhere around 10 or 20 per cent) which does not affect the usefulness of the method for practical purposes.

As the milk surrounding the counter does not provide much shielding against external gamma-radiation, external shielding is desirable if the gamma-background amounts to more than about 0.05 mr/hr.

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Angular Response of a Partially Shielded Cylindrical Dosimeter

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MANY cylindrical dosimeters have very thin walls and are partially shielded by a reinforcing structure. If tops and sides of the dosimeter are unequally shielded the response of the dosimeter will depend upon the angle between the plane of the cylinder base and the direction of the incident radiation. For instance, a cylinder with opaque sides will admit all radiation perpendicular to the plane of the base and none parallel to it. If the volume irradiated through the base and the volume irradiated through the sides are known separately as functions of the angle of incidence, the total irradiated volume of a shielded dosimeter may be determined as a function of the angle of incidence.

In Fig. 1, θ is the angle between the direction of the incident radiation and the plane of the base of the cylinder, and ϕ is an angle in the plane of the base between a radial line through the center of the base and the reference direction. A plane, at a distance $x = p \sin \phi$ from the axis of the cylinder and parallel to the reference direction, forms the base of an infinitesimal rectangular volume. It is convenient to consider two cases.

Case I

$$\cos \phi > \cos \phi_c = h/2p \tan \theta$$

or $\quad \phi < \phi_c$

Case II

$$\cos \phi < \cos \phi_c$$

or $\quad \phi > \phi_c$

Both cases are illustrated in Fig. 1.

The infinitesimal rectangular volumes have been

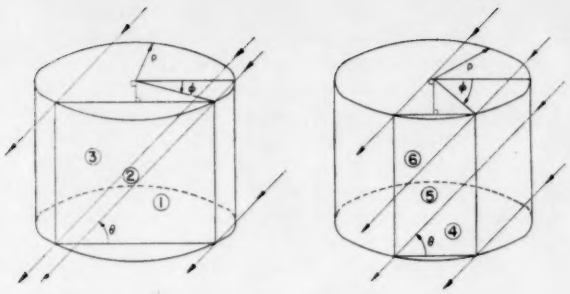

FIG. 1. Case I: $\phi < \phi_c$, Case II: $\phi > \phi_c$.

divided into three regions in both cases. In Case I it is obvious that region (1) will be irradiated through the side while regions (2) and (3) will be irradiated through the top. Similarly in Case II regions (4) and (5) will be irradiated through the side, while region (6) will be irradiated through the top.

In Case I the infinitesimal volume of region (1) is

$$dV_1 = \frac{h^2 p \cos\phi d\phi}{2 \tan\theta}$$

the volume of region (2) is

$$dV_2 = [2p \cos\phi - h/\tan\theta] hp \cos\phi d\phi$$

and the volume of region (3) is

$$dV_3 = dV_1$$

In Case II the infinitesimal volume of region (4) is

$$dV_4 = 2p^3 \tan\theta \cos^3\phi d\phi$$

the volume of region (5) is

$$dV_5 = 2p^2 \cos^2\phi d\phi\,[h - 2p \cos\phi \tan\theta]$$

and the volume of region (6) is

$$dV_6 = dV_4$$

Now the volume irradiated through the top will be

$$V_{top} = 2\int_0^{\phi_c} \{dV_2 + dV_3\} + 2\int_{\phi_c}^{\pi/2} dV_6$$

and that through the side will be

$$V_{side} = 2\int_0^{\phi_c} dV_1 + 2\int_{\phi_c}^{\pi/2} \{dV_4 + dV_5\}$$

The solutions of these integrals are

$$\frac{V_{top}}{\pi p^2 h} = \frac{2\phi_c}{\pi} + \frac{\sin 2\phi_c}{\pi} - (h/p)\frac{\sin\phi_c}{\pi \tan\theta} + \frac{4 \tan\theta}{\pi(h/p)}\left[2/3 - \sin\phi_c + \frac{\sin^3\phi_c}{3}\right]$$

and

$$\frac{V_{side}}{\pi p^2 h} = 1 - \frac{V_{top}}{\pi p^2 h}$$

where $\phi_c = \cos^{-1}[(h/p)/2 \tan\theta]$.

Note the dependence of the fractional irradiated volumes on the parameter (h/p), the "tallness" of the cylinder. The functions

$$\frac{V_{top}}{\pi p^2 h}[\theta,\ (h/p)] \text{ and } \frac{V_{side}}{\pi p^2 h}[\theta,\ (h/p)]$$

have been plotted (Fig. 2) for several values of the parameter (h/p) including $(h/p) = 0.521$, which

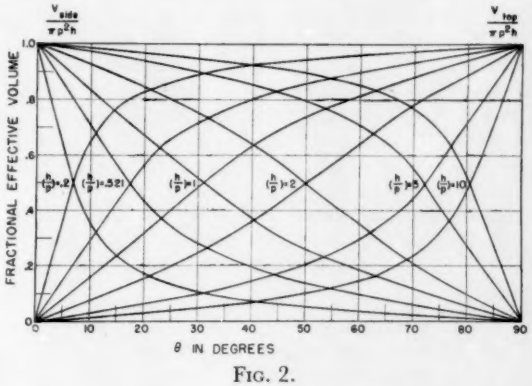

FIG. 2.

applies to a beta-dosimeter developed at ORNL.[1]

If A_{top} is the unshielded area of the top and A_{side} is the unshielded area of the side, then the effective irradiated volume $V_{eff}(\theta)$ at an angle of incidence θ, is

$$V_{eff}(\theta) = \frac{A_{top}}{\pi p^2} V_{top}(\theta) + \frac{A_{side}}{2\pi ph} V_{side}(\theta)$$

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A Portable γ-Dosimeter Providing Audible Warning

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1. *Introduction*

The routine monitoring of gamma-radiation exposure is carried out at the Chalk River Project by the use of film dosimeters and, in certain instances, with pencil chambers. Both of these detectors provide useful information for the administrative control of long-term radiation dosage but they fail to prevent the accumulation of high doses which sometimes occur with unexpected changing fields. This is so because the information is obtained only when the films are processed or when the pencil chambers are read.

For this reason, a dosimeter is needed which alarms when a predetermined dose has been exceeded. It has been decided at AECL that this is best provided by a device which sounds an audible alarm when a chosen dose has been accumulated. A survey of commercial instruments of this type showed that it was necessary to develop a personal-warning dosimeter in order to satisfy local requirements. Such an instrument should:

(a) be clearly audible when worn with protective clothing in noisy surroundings,
(b) not require any auxiliary equipment,
(c) be easy to decontaminate,
(d) give a continuous audible indication of normal operation.

An instrument using similar circuitry[1] to that described in this report has been developed, but it does not meet the first two requirements.

Since 300 mr is the maximum permissible weekly dose it was decided to set the warning level of the dosimeter at 200 mr. This leaves a margin of 100 mr for error or for the dose acquired in the rest of the week.

2. *Description*

The dose is measured by accumulating the charge, obtained from a saturated ion chamber, in a low-leakage condenser. When the voltage on the condenser rises to a set value, a regenerative circuit is triggered. This circuit then switches power to an oscillator driving a small loud-speaker.

Fig. 1 is a circuit diagram of the dosimeter. To set the dosimeter, switch S_2 is closed temporarily, thus short-circuiting the Zener diode CR_1 and bringing the cathode potential of the electrometer tube down to that of ground; consequently, the electrometer grid swings negative with respect to ground by about 1 V. This small negative grid bias permits a substantial current to flow to the anode; hence its potential falls. As a result, transistor Q_2 is turned on so that its collector rises in potential. Since the collector is connected to the screen grid of the electrometer tube, the anode current rises regeneratively until the trigger circuit, which comprises the tube and the transistor Q_2, is fully turned on. Consequently, a current of 3 mA flows into the base of the transistor Q_1, which permits continuous oscillation, resulting in a steady audible sound from the loudspeaker.

When the switch S_2 is permitted to open again, the dosimeter is ready to measure dose and the following sequence occurs. The cathode of the electrometer tube rises to the Zener voltage (around 9 V) of the diode CR_1 and the anode current is immediately cut off since the grid remains below ground potential. The trigger action then turns the circuit back to its quiescent condition and the current to the oscillator is cut off. The oscillator then changes its mode of operation from continuous oscillation to short pulses. This pulsed operation occurs because direct current is permitted to flow in the oscillator circuit by way of a large resistance connected between collector and base of transistor Q_1. The resulting audible signal gives an indication that the dosimeter is operative.

If the ion chamber is placed in a radiation field, the current from the collecting electrode will charge the integrating capacitor which is connected between the electrometer-tube control grid and ground. As a result, the grid potential rises until it is within about 1 V of the cathode when the trigger circuit fires in the way described previously, causing the alarm to be sounded. The instrument must then be reset.

Fig. 2 shows the mechanical arrangement of the dosimeter.

3. *Performance*

The dose at which the trigger circuit fires is determined by the chamber volume, the Zener voltage of the diode CR_1 and the total input capacity. The influence on this level, of the tube characteristics and the battery voltage, is much reduced by the use of a diode with a Zener voltage of around 9.0 V. The

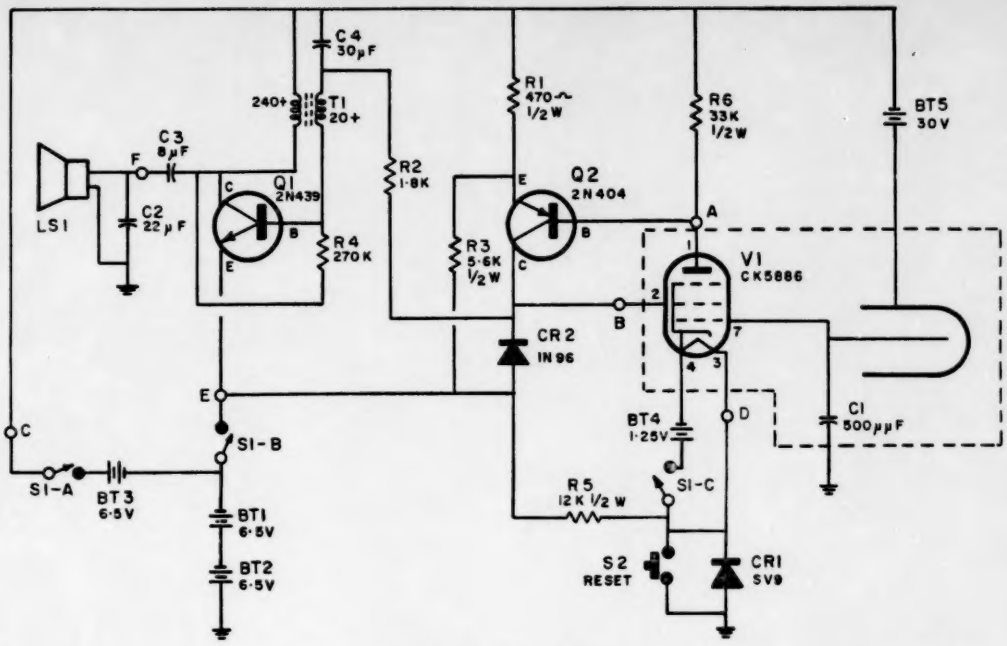

FIG. 1. Circuit diagram.

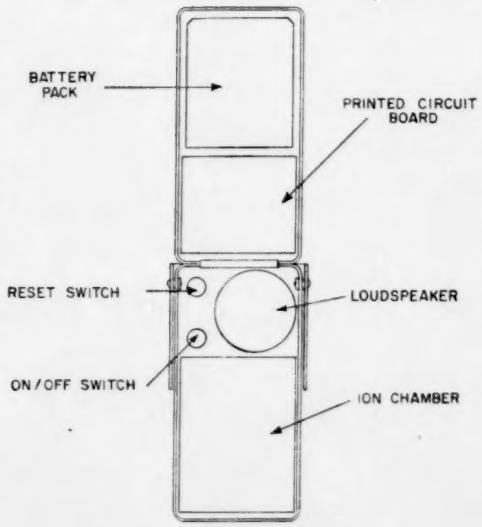

FIG. 2. Mechanical arrangement.

error due to all these factors should not exceed 20 per cent.

The most serious limitation on accuracy in this kind of instrument is that imposed by dose-rate dependence. At very high dose rates the sensitivity falls off due to the saturation characteristics of the ion chamber. At very low dose rates, leakages to or from the integrating capacitor may cause error.

Calculations of dose-rate dependence (Appendices I and II) show that the response of the dosimeter should vary less than ±20 per cent over the range of dose rates 80 mr/hr to 1600 r/hr. This dependence should be observed when the leakage resistances have their minimum values permitted by specification. Bearing in mind the purpose of the instrument, which is to prevent the accumulation of large doses in short periods, this performance is satisfactory. At 1600 r/hr, 200 mr is accumulated in 0.45 sec and, since it is not possible to do a useful amount of work in periods shorter than this, extension of dose-rate dependence to higher fields is unnecessary. Dose rates less than 80 mr/hr are not likely to result in serious overexposures because at that rate it would take $3\frac{3}{4}$ hr to acquire a maximum permissible dose for one week (300 mr). In any case, the dosimeter is not used in lower fields. An accurate response at lower dose rates could be attained only at the cost of increasing size or specifying higher leakage resistances (Appendix II). The predicted and measured dose-rate dependences are shown in Fig. 3.

Another feature of the design to which special attention has been paid is the output of sound from

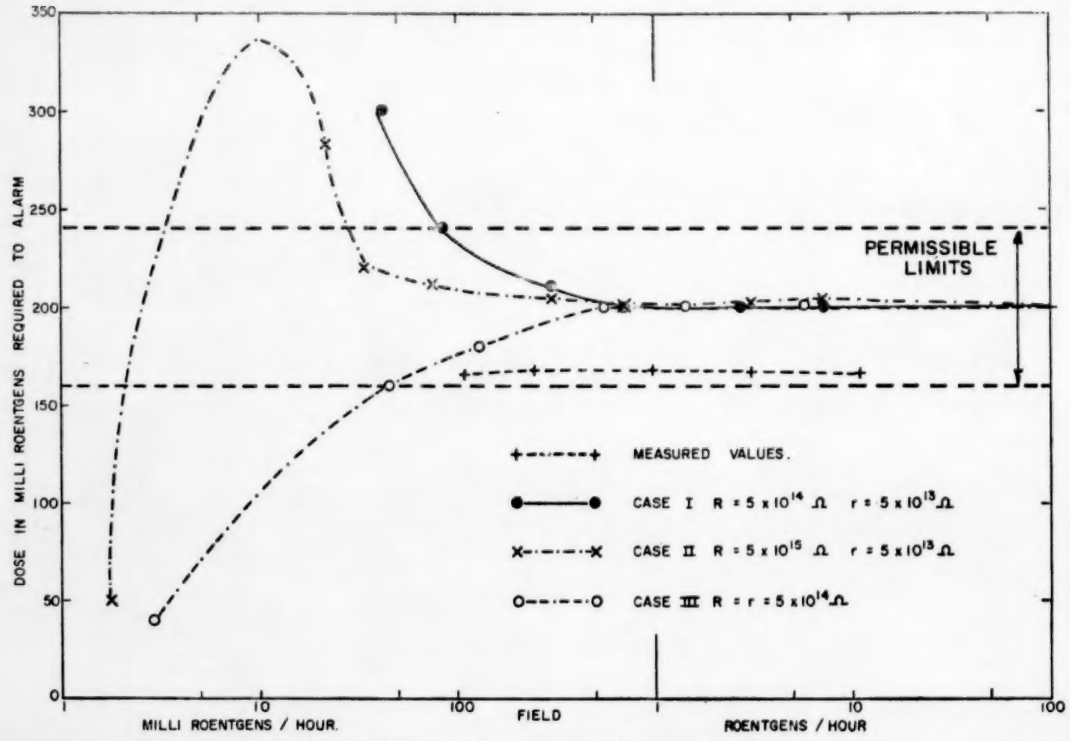

FIG. 3. Calculated and measured dose-rate dependence.

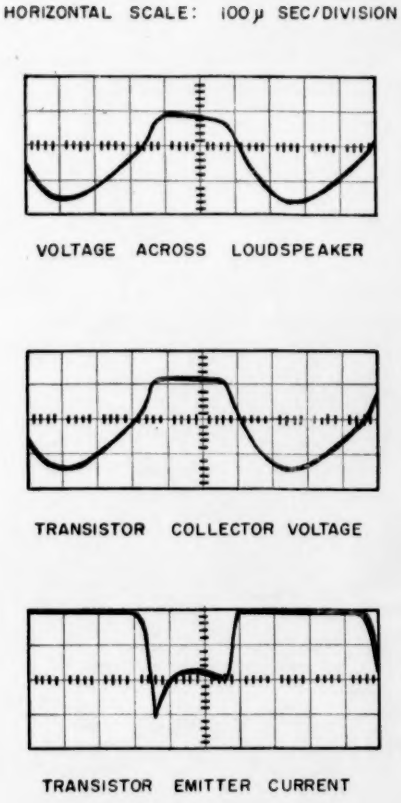

FIG. 4. Oscillator wave forms.

the speaker. As mentioned before, this seems to be too low in some instruments of this kind. In the present dosimeter, about 0.1 W of electric power applied to the speaker provides a clearly audible sound. Very little power is dissipated elsewhere in the circuit because the transistor is operated in its switching mode and because the saturation voltage is low compared to the peak voltage applied to the speaker. This is illustrated in Fig. 4. The center wave form is that of the collector of voltage transistor Q_1 while the lowest wave form shows the emitter current. The top one shows the voltage across the speaker. The phase relationship between emitter current and collector voltage shows that most of the power in the circuit is dissipated in the loud speaker.

Since the dosimeter may be worn with protective clothing, care has been taken to prevent the muffling of the loudspeaker. The acoustic output is enhanced by a cover which is placed over the loudspeaker aperture keeping any clothing away.

While the dosimeter is in the quiescent condition, the wave forms are not altered appreciably but the "on" time of the pulses is small compared with the "off" time, thus permitting continuous operation of the circuit for more than 200 hr. The battery which supplies power to the oscillator runs down more rapidly than the other batteries. Consequently, if the oscillator is functioning, adequate power is available to the rest of the circuit. If all batteries are changed when the oscillator ceases to function, there is little danger of the dosimeter becoming inoperative due to battery run-down without the user being made aware of it.

Acknowledgements—Measurements used in the development of the dosimeter were made by A. Ohno and D. Robertson. The Product Engineering Section, Engineering Design Branch contributed the mechanical design of the instrument.

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APPENDICES

I. *Saturation of the ion chamber*

The fraction[2] of the current in an ion chamber which is lost by recombination is:

$$\frac{\alpha N_0 d^4 p^2}{\mu_+ \mu_- v^2}$$

where α is the recombination coefficient, N_0 is the rate of production of ions in a unit volume, p is the pressure in mm of mercury, $\mu_{\pm}$ is the mobility of the positive/negative ions, v is the voltage across the electrode gap and d is the electrode gap.

The fraction[2] of current lost by diffusion is:

$$\frac{\varepsilon K T}{ev}$$

where ε is the ratio of mean energy of ions with and without electric field, K is Boltzmann's constant, T is the absolute temperature, v is the voltage across the electrode gap.

Only the loss due to recombination depends on N_0 and therefore only this loss contributes to the dose-rate dependence.

The first expression may be used to compute the fractional loss at 1000 r/hr. This turns out to give an error of 12 per cent in measurement for a chamber of the size and shape used in the dosimeter.

II. *The effects of leakages to and from the integrating capacitor*

Fig. 5 shows the equivalent circuit of the leakages which can cause error in the operation of the dosimeter. From this diagram, it can be seen that:

$$v = R\left(i[t] + \frac{dq}{dt} - I\right) + \frac{q}{c}$$

where v is the voltage (50 V) between the positive ion-chamber electrode and ground.

$$ri[t] = \frac{q}{c}$$

and eliminating $i[t]$

$$\frac{q}{c}\left(\frac{R}{r} + 1\right) + R\frac{dq}{dt} - (RI + v) = 0$$

The initial condition is set when the reset switch removes the charge from the integrating capacitor, i.e. $q = 0$, $t = 0$.

Then the required solution of the equation is:

$$q = \frac{c(v + RI)}{R/r + 1}\left[1 - \exp\left\{\frac{-(R/r + 1)t}{CR}\right\}\right]$$

The dose D received by the ion chamber is proportional to the product of the current and time, i.e. $D = KIt$.

If the voltage of the capacitor at which the trigger circuit fires is ϕ then:

$$D = \left[\frac{\phi(R/r + 1)}{1 - \exp\left[-(R/r + 1)t/CR\right]} - v\right]\frac{Kt}{R}$$

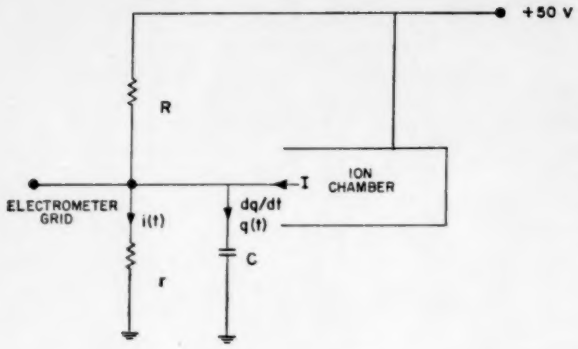

FIG. 5. Diagram of ion-chamber leakages: R is the total leakage resistance between the electrodes of the ion chamber; r is the total leakage resistance between the electrometer grid and ground; C is the total integrating capacity between grid and ground; $q(t)$ is the charge on the capacity C.

If $R = r = \infty$, then $D = KC\phi$ and then the ratio of true to apparent dose is:

$$\left[\frac{\phi(R/r + 1)}{1 - \exp\left[-(R/r + 1)t/CR\right]} - v\right]\frac{t}{CR\phi}$$

This expression was used to compute the dose-rate dependence shown in Fig. 3.

Three cases have been computed. In case (1), the values of R and r are the minimum ones which are allowed, i.e. $R = 5 \times 10^{14}\ \Omega$, $r = 5 \times 10^{13}\ \Omega$. In case (2), $R = 5 \times 10^{15}\ \Omega$, $r = 5 \times 10^{13}\ \Omega$. In case (3), $R = r = 10^{14}\ \Omega$.

As can be seen, case (2) is the most serious, but the readings are within the permissible range if the dose rate exceeds 80 mr/hr. Values taken from measurements on the prototype are much better than those from either of the three predicted curves. Presumably, the leakage resistances are considerably more than the minimum values specified.

Lead Suspension for Low-energy Radiation Shielding*

(*Received* 14 *January* 1960)

THE trend toward downward modification of maximum permissible limits for exposure to ionizing radiation, coupled with the philosophy that exposure should be held to the least practicable value, has suggested the desirability of developing a shield that can be applied to existing intermediate and low-energy X-ray tube housings that fail to comply with current recommendations regarding leakage radiation transmitted through the tube housing. Such a shield would also find application in the additional shielding of darkrooms and radiation enclosures.

To this end, a lead suspension has been devised, which is easily applied by brush or spray-gun, forms a continuous surface, conforms to existing contours, and is capable of being finished to match existing decor.

Very finely divided elemental lead† is suspended in Plicote Laboratory Paint‡ and thinned with Plicote Thinner.§ This paint contains a cyclicized derivative of synthetic rubber, and is characterized by resistance to acids and alkalis and to heat, up to 350° F, and by some degree of flexibility. It is soluble in coal tar hydrocarbons, esters and ketones.

The paint adheres well to steel and brass; preparation of the surface with a primer is not required. In the case of black Plicote Paint, which was used in this study, the unfinished coat is the color of the Plicote Paint, and when sanded it appears as a homogeneous surface of bright lead with no binder visible. When dry, the lead–Plicote surface may be finished with paint or enamel to match the existing color scheme of the equipment.

When distributed over 1000 cm², 113.4 g of lead will provide an average thickness of 0.1 mm. The non-uniformity of the lead layer will vary with the manner of application (i.e. brush or spray), the number of coats and the skill of the worker. Investigation of 1 cm² sections of the lead layer indicate that application by brush leads to about twice the degree of non-uniformity achieved by spraying; i.e. brushing as described gives a thickness of 0.1 ± 0.015 mm and spraying 0.1 ± 0.008 mm. On this basis, it is suggested that a safety margin of 25 per cent be incorporated in calculations of the amount of

* Based on work performed under Contract AT-30-1-GEN-70 for the U.S. Atomic Energy Commission.

† Lead Metal, Finest Powder, Fisher Scientific Company, Cat. No. L-29.

‡ Plicote Laboratory Paint, Fisher Cat. No. 13-312.

§ Plicote Thinner, Fisher Cat. No. 13-312-15.

suspension required to cover a given area with a given thickness of lead.

Table 1 below indicates the suggested constitution of the suspension. The figures given are for coverage of 1000 cm^2 of surface with 0.1 mm Pb, including the 25 per cent safety factor:

Table 1

	Brush	Spray
Lead	142 g	142 g
Plicote Paint	30 ml	30 ml
Plicote Thinner	4 ml	30 ml

The suspension has been successfully sprayed with the following guns:

(1) Roxon Volumaire R.200 Portable Spray Unit.*

(2) A De Vilbiss spray unit (fixed installation), type MBC, with No. 30 head, AV-15-EX feed, operated at a pressure of 70 lb/in^2.†

The sprayed suspension dries with a finely stippled surface that does not require sanding prior to finishing with paint or enamel. Application by brush results in a striated surface that is best sanded smooth before finishing. Alternatively, a first finishing coat may be applied before sanding, resulting in less loss of lead, although, in any case, this is slight.

The amount of additional shielding required to bring diagnostic tube housings into conformance with current recommendations[1] is relatively small. For example, for 75 kV machines, a lead layer 0.6 mm thick will attenuate the leakage radiation to about 10 per cent.[2]

The desired thickness should be built up in steps of not more than 0.1 mm, with drying time of at least $\frac{1}{2}$ hr between coats. To maintain a uniform suspension, and to prevent jamming of the spray-gun, the suspension must be constantly agitated during application, because of the tendency of the lead particles to settle rapidly.

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† De Vilbiss Co., 10 Stern Avenue, Springfield, N.J.

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NEWS

News Editor: J. J. FITZGERALD

Radiological Health Curriculum Symposium held at Princeton

A Radiological Health Curriculum Symposium under the sponsorship of the Public Health Service was held at Princeton, New Jersey, on 2, 3 and 4 August 1960. Approximately 130 university personnel and other interested personnel were invited to the symposium to present papers and to participate in the formation of criteria and methods whereby the rapidly increasing number of health physicists or radiation safety physicists may be effectively educated in accordance with the national needs, as indicated in the last issue of Health Physics News.

Survey shows no added radioactivity off Boston Harbor

A survey of a site off Boston Harbor formerly used for the disposal of limited quantities of packaged radioactive wastes has revealed no radioactivity attributable to the disposal operations.

Samples of water, sediments and marine organisms living in the area were collected by the U.S. Coast and Geodetic Survey and were analyzed for radioactivity by the U.S. Public Health Service at its Robert A. Taft Sanitary Engineering Center in Cincinnati. The radioactivity detected was found to be in the same range as that of background activity at other ocean locations wherein radioactive wastes have not been disposed.

The site was used under AEC authorization and license from 1952 to August, 1959, by Crossroads Marine Disposal Corporation of Boston for the disposal of low-activity packaged radioactive wastes which had a total of 2434 c at the time of disposal. The former site is an area 2 miles in diameter located 15 miles off Boston Harbor in Massachusetts Bay.

The license of the company was amended by the AEC in August, 1959, to require the firm to carry out its operations in deep water (1000 fathoms) off the continental shelf. No further use of the area off Boston Harbor for disposal purposes is contemplated. The Commission's present policy is to require that wastes be disposed of in water at least 1000 fathoms deep and the Commission is not contemplating any change in that policy.

International Standards Organization meets in Geneva

Representatives of the Sectional Committees N-6 (Reactor Safety) and N-7 (Radiation Protection) of the American Standards Association (ASA) participated in the ISO Technical Committee 85 Meeting in Geneva during the week of 23 May 1960. The ASA N-6 representatives had prepared documents on site selection criteria, dynamics characteristics as related to reactor coolant, operator qualification and containment. The ASA N-7 representatives presented a draft of a "Standard for Uranium Mines and Mills". The Uranium Mine and Mills standard in draft form has been approved by The Sectional Committee (N-7) and is being submitted to the sponsors for their approval. Only conditional approval of the above drafts was requested, since there are differences between the draft and NCRP levels for radon daughter products and since the draft does not include the subject of waste disposal practices or a standard for thorium mines and mills. The representatives of N-7, requested the approval of the present draft because uranium miners and concentrators have expressed a great need at this time for these working standards or guide lines.

National radiological advisory service recommended by British committee

A committee of the United Kingdom Atomic Energy Authority with Sir DOUGLAS VEALE, Chairman, has submitted a report to the AEA on training in radiological health and safety. In essence, the report states that full time health and safety staffs for many organizations were an impracticality for the companies and would mean a wasteful use of skilled manpower. The Committee reports: "We, therefore, came to the conclusion that the only way of providing the services of health and safety specialists to those organizations not requiring them full time would be to establish a national radiological advisory service." In summary, the Committee made in part the following recommendations.

1. Establish a national radiological advisory committee.
2. Do not establish a national training center.
3. Existing provisions for formal training of safeguard staff are adequate.
4. Existing and projected university post-graduate courses are adequate and when fully developed will meet estimated demand.

5. Other universities should be encouraged to accept students for degrees in disciplines relevant to radiological health and safety.
6. With respect to the medical profession, the curriculum for the Diploma in Public Health should include a course of instruction in radiological health and safety.
7. Medical schools should consider the desirability of including in the basic medical curriculum some instruction in the use and hazards of ionizing radiations.
8. In the area of public information and communications, the Authority should explore the possibility of publishing books and pamphlets, of showing films and the production of radio and television programs on the subject of radiological health and safety.
9. The Authority should not provide general courses which can be given equally well by universities and colleges.

International regulations on "Safety of Life at Sea" adopted

The International Convention of "Safety of Life at Sea" (SOLAS) met in London and the following regulations were adopted.

1. The government of the country in which the nuclear ship is registered (registering government) has the responsibility for approving the design, construction and standards of inspection and assembly of reactor installations, and also assumes responsibility for insuring adequate radiation protection of "the crew, passengers, public . . . waterways, food and water resources".
2. A "Safety Assessment" will be prepared on each nuclear ship and approved by the registering government. This report will be sent to the governments of countries the nuclear ship intends to visit sufficiently in advance so that the nation visited may independently evaluate the safety of the ship.
3. A detailed "Operating Manual" covering "all matters relating to the operation of the nuclear power plant and having an important bearing on safety" shall be prepared and approved by the registering government.
4. Surveys of nuclear ships shall be conducted at least once a year, on the basis of which the registering government will issue a "Nuclear Ship Safety Certificate".
5. Prior to entering port the appropriate port authorities are given "special control" (i.e. inspection rights) over nuclear ships in order to verify that a valid Nuclear Ship Safety Certificate is on board and "that there are no unreasonable radiation or other hazards at sea or in port".

PHS issues first monthly report on "Radiation Health Data"

The first issue (29 April) of a new monthly technical publication which presents the concentration of radioactive materials in air, water and milk was made available by the U.S. Public Health Service, Division of Radiological Health. Initially, it is anticipated that most of the materials contained within the report will be in the form of raw data. With the accumulation of data and with more knowledge of the significance of low-level exposures it is felt that analyses and interpretations of the data will become more feasible and appropriate.

The data for the new publication are compiled from many sources by the Division of Radiological Health with the assistance of a board of editorial advisors representing the Departments of Health, Education and Welfare; Defense, Agriculture; Commerce; and the Atomic Energy Commission.

The new publication is an outgrowth of a directive by the President in August (1959) that the Department of Health, Education and Welfare intensify its radiological health efforts. The publication, "Radiation Health Data" may be purchased from the office of Technical Services, Department of Commerce.

PHS survey on the effects of high-level background activity on residents of San Juan County

A study of the effects of environmental radiation on the health of large population groups will be conducted by the U.S. Public Health Service. This study will be initiated in San Juan County, New Mexico, with the co-operation of the State Health Department. The study will be extended to the adjacent areas along the Animas and San Juan and to other parts of the country. In a survey which began in 1958, the Animas River was found to contain concentrations, from natural sources, of radioactive materials which exceeded maximum permissible levels. Approximately 100 families will be examined. These examinations will involve detailed medical and laboratory analyses.

Additional studies are being conducted in the northeastern portion of the U.S.A. to ascertain the significance of relatively high levels of natural activity in sources of water-supply on the large populations over long periods of time.

AEC designated Oak Ridge and Idaho Falls as burial sites

Two interim land burial sites were established by the AEC for the disposal of solid packages of radioactive wastes generated by AEC licensees. Wastes must be packaged as required by the Interstate Commerce Commission regulations. The licensees will pay the transportation costs. Charges for burial at Oak Ridge or Idaho Falls will be at the rate of 70 cents per cubic foot with a minimum charge of $21.00 for 30 cubic feet of packaged waste or less.

AEC rights to regulate waste disposal upheld

The right of a waste disposal organization, the Coastwise Marine Disposal Company, to conduct its business in accordance with a proper AEC license was upheld by the Los Angeles Superior Court. The court stated: "The Atomic Energy Commission has been granted full power and sole responsibility for regulating the disposition of radioactive waste and protection of the public health thereby. Congress (in the Atomic Energy Act of 1954, as amended) has decreed that the Commission may not delegate regulation of such waste disposal even to a state. Therefore, the City of Long Beach is without power unreasonably to interfere with those activities of plaintiff (Robert Boswell, Owner of Coastwise Marine) for which he is licensed by the AEC."

In addition, however, the court said: "State and municipal agencies may regulate the activities of an (AEC) licensee to such extent as does not unreasonably interfere with/or frustrate the national objective committed to the exclusive jurisdiction of that Commission."

AEC report on 12th criticality accident

A report on the 12th criticality accident was made on 8 April 1960, in the "Serious Accident" publications of the Safety and Fire Protection Branch, Office of Health and Safety, U.S. A.E.C. The report states that a nuclear incident (estimated 4×10^{19} fissions) occurred in a process equipment waste collection tank at one of the AEC chemical processing plants. Radiation alarms in the plant were set off by the resulting release of airborne radioactivity. Twenty-one shift workers and security personnel on duty evacuated the process building and the surrounding area of high radioactivity.

Available evidence indicates the critical condition followed an accidental transfer of concentrated uranyl nitrate solution from geometrically safe storage tanks to a waste collection tank through a line normally used to transfer decontaminating solutions. Abnormal and unexpected siphon action provided the most likely mechanism by which the transfer took place.

The waste collection tanks were located approximately 15 ft below ground with a 4 in. thick concrete deck over the vessels. The shielding effectively prevented excessive radiation from reaching the operating area during the period of inadvertent criticality within the waste collection tank.

Limited visual inspections and tests indicate that no significant property damage or loss resulted beyond the approximately $60,000 to recover the contaminated uranium solution induced by the incident.

Of the twenty-one personnel directly involved in this incident, seven received significant external exposure to radiation. Of the seven, none received a year's maximum permissible "whole body" exposure of penetrating radiation. Two exceeded the year's maximum permissible exposure to the skin. As reported, these individual external exposures were 50 rems and 32 rems. No medical treatment was required for the twenty-one personnel involved.

Court action in radiation cases

The jury was dismissed last month as hopelessly deadlocked in the suit of Benjamin Zawacki, electrician, against Sam Tour and Company, Inc., for negligence in a 1956 cobalt-60 exposure case. Zawacki at this time pocketed a cobalt-60 capsule which was being used on a construction job for radiographic inspection work, and it was not recovered until the following morning. Zawacki alleged health impairment and mental anguish. Date for a new trial has not been set.

In another radiation case, Arthur W. Hermann brought a million dollar damage suit against the United States Radium Corporation for cancer allegedly caused by exposure to radiation from radium while in the company's employ between 1918 and 1922.

Radiation incident at Oak Ridge

A faulty ventilation system at the Solid State Building at the Oak Ridge National Laboratory led to a release of radioactive material and resultant evacuation of the building. Eight persons were slightly contaminated by the material, which consisted of graphite containing cerium and strontium from a fuel element which was being examined in a process cell. Clean-up of the building was at first expected to take two weeks but was complicated by the fact that material had got into the building's air conditioning system. No activity was released from the building.

Atomic energy safety record exceeds industrial safety record

The Assistant General Manager of the AEC, Dwight A. Ink, on 11 July, presented safety awards to the Sandia Corporation for the records that their Livermore Laboratory personnel have achieved. The Sandia Laboratory in Livermore has passed the impressive mark of 12.5 million accident-free man-hours. During the presentation, the following comments were made in part with respect to the atomic energy safety record.

1. "Less than 1 per cent of the fatalities in the atomic energy industry involved radiation. The great majority of fatalities in this field are of the usual industrial type from falls, electric shock and traffic."
2. "Among the 114,000 workers in the AEC laboratories and plants which are operated by academic and industrial contractors, we find during the first five months of this year an overall frequency rate of only 1.69 injuries per million man-hours. The most recent figure available from the National Safety Council reflects an average rate for all types of industry nearly four times as high. We also find that the average accident in our atomic energy facilities is less severe than the average in other types of industries."

Massachusetts Commission on Atomic Energy publishes newsletter

The first issue of the newsletter "Nuclear Notes" has been distributed by the Massachusetts Commission on Atomic Energy. The newsletter will contain brief and timely statements on the developments in the atomic energy industry such that it may keep the Commonwealth adequately informed.

Health physics activities in France—F. DUHAMEL

Nuclear energy in France. The Atomic Energy Commission has envisioned the creation of a new atomic energy center for the study of underground explosions. The new site will be located at Corsica in the main part of Argentelle which is approximately 11 miles south of Calvi.

In the laboratories of the new center, the specialists on the control and detection of radiation will pursue their studies under physician, YVES ROCARD, a professor at the Normale School of Paris.

Radioprotection Society in France. An association called the Radioprotection Society has recently been formed in France.

This Society has as an objective for the group of specialists the development of the functions of radioprotection organizations: the exchange and diffusion of the results obtained and making known principles and methods of radioprotection.

The principle disciplines involved are the following:

Detection and Dosimetry
Prevention and Safety
Radiation Engineering and Technology
Treatment of Radioactive Waste
Geophysics and Ecology
Radiobiology (plant and animal)
Radiopathology and Radiotoxicology
Organization and Legislation
Responsibility and Education

This Society is directed by Dr. HENRY JAMMET, Chief of the Atomic Hygiene Service and of radiopathology for the Atomic Energy Commission. Professor BUGNARD, Director of the National Institute of Hygiene is the President of Honor.

Health physics activities in Italy—ENRICO VIA

Sponsored by the National Committee for Nuclear Research, the "Third Course for Medical Applications of Radioisotopes and High Energies", under Professor L. TURANO, will be held in Rome, at the Institute of Radiology.

The course has a duration of three months. Special importance will be given to the subject of General Pathology of Radiations; Hygiene of Radiations; and Health Physics and Legislation.

A course of Nuclear Law organized by the International Center of Studies and Documentation of the European Communities will be held in Milan.

Health physics activities in Sweden—BO LINDELL

In Sweden there is a very close link between health physics and medical physics activities. The *Radiation Protection Board*, consisting of Chairman, Director of Institute of Radiophysics and five other members, licenses all work involving use of radioactive substances, X-ray equipment and other technical installations emitting ionizing radiation, as well as trade, import, acquisition, possession and transfer of radioactive substances.

The Institute of Radiophysics at Stockholm, acts as the executive agent of the Board. The Institute is responsible for supervision of all establishments where exposures to ionizing radiation occurs or where radioactive substances are stored. However, it is also a research institute, and the Director is a Professor of Radiophysics. The Institute has one department for clinical radiophysics working for the cancer clinic "Radiomhemmet".

There are departments of medical radiophysics at the University of Lund and within the Medical Faculty in Gothenburg.

Within the Atomic Energy field, the *Atomic Energy Board* is the co-ordinating body for all matters connected with the development in the nuclear energy field. The State holds 4/7 of the shares and a minority is owned by private industry. The Company is running a number of reactors, namely:

R0	50 W in Studsvik	(1959)	(Research)
R1	1 MW in Stockholm	(1954)	(Research)
R2	30 MW in Studsvik	(1960)	(Material Testing)
R2	100 KW in Studsvik	(1960)	(Research)
R3/Adam	65 MW in Agesta (under construction)		(Heat and Power)

The Atomic Energy Company has special Health Physics groups, although the Institute of Radiophysics has to approve the protection measures and make frequent inspections.

Also the *State Power Board* (which is planning the R4/Eve reactor east of Norrkoping) has an interest in the nuclear energy field and has a health physics group.

There is a national health physicist organization called the *Swedish Association of Health and Hospital Physicists*, having approximately 40 members who have responsibilities in the fields of representing and inspecting atomic energy and clinical radiophysics. Another organization, the *Medical Physics and Techniques Society* (having both physicists and physicians as members), is partly in opposition to the radiation physicists' claim of representing medical physics in a broader sense. This conflict seems to reflect trends existing also on the international level.

During the last few years, "radiophysics" has been a subject that can be included in postgraduate studies.

Health physics activities in Ireland—LAWRENCE COLLERY

A three-day course on the Properties and Hazards of Nuclear Radiations was given in the Physics Department, University College, Dublin, on 6, 7 and 8 April 1960.

Over one hundred people attended, including Medical Officers, Veterinary Officers, Engineers, Surveyors, Fire Chiefs and Civil Defense Officers, from almost every Local Authority in the twenty-six Counties. Members of various Government Departments also attended as well as representatives from other Public Bodies, the Army and industry. The program was as follows:

Professor M. A. HOGAN, Professor of Civil Engineering:
- Disposal of Radioactive Wastes
- Removal of Radioactivity from Water Supplies

Professor T. MURPHY, Professor of Social and Preventive Medicine:
- Atomic Energy and Mental Health

Professor T. E. NEVIN, Professor of Experimental Physics:
- Atomic and Nuclear Structure of the emitted radiations
- Radioactivity and the properties
- Neutrons and the reactions they produce
- Radioisotopes
- Fission
- Reactors and reactor hazards
- Production of radioisotopes
- Bombs

Mr. F. O. FOGHLUDHA, Physicist, St. Luke's Hospital:
- Radiation units
- Radiation detection
- Principles of protection
- Decontamination

Dr. MALACHY POWELL, Medical Inspector, Department of Health:
- Biological Principles
- Absorption of radiation
- Delayed effects
- Genetic effects
- Metabolism of radioactive substances
- Strontium and other substances in food chain
- Organization of Medical Service

The last period of the course was devoted to discussion during which those attending were invited to ask questions relating to their individual problems.

Health physics activities in the United Kingdom—H. J. DUNSTER

An International Symposium on Inhaled Particles and Vapours* was organized by the British Occupational Hygiene Society and was held at Oxford, England, from 29 March to 1 April 1960. The purpose of the symposium was to discuss the physical, chemical and physiological factors governing the entry of harmful substances into the body by way of the respiratory system.

Thirty-eight papers were presented, seven of them being concerned with the inhalation of radioactive particles; it was interesting to note that all the authors of these papers came from the U.S.A. Authors of papers on other subjects hailed from Great Britain, U.S.A., Germany, France, Sweden, Italy, Israel and Czechoslavakia. The symposium provided a considerable opportunity for workers with radioactive inhaled particles and vapours to compare notes

* C. N. DAVIES (Editor), *Ann. Occ. Hyg.* In press (1960).

with those working on similar problems in the non-radioactive field.

Release of Statements by the National Committee on Radiation Protection and Measurements (NCRP)

(1) At its meeting in November 1959 the Executive Committee agreed that the NCRP should make a statement with regard to the maximum permissible dose from television receivers. Such a statement has been prepared and voted upon by the full committee.

The following position has been adopted by the NCRP:

During the past years members of the NCRP have investigated the emission of X-rays from television receivers. From a genetic point of view even sources of minute radiation are of significance if they affect a large number of people. X-rays emitted by home television sets are, therefore, of interest because of the high percentage of the population involved. In order to insure that the television contribution to the population gonad dose will be only a small fraction of that due to natural background radiation, the NCRP recommends that the exposure dose rate at any readily accessible point 5 cm from the surface of any home television receiver shall not exceed 0.5 mr/hr under normal operating conditions.

Laboratory and field measurements* have shown that with this maximum permissible exposure level the television contribution to the gonad dose at the usual viewing distances will be considerably less than 5 per cent of that due to the average natural background radiation. Most of the present television receivers already meet this requirement with a high factor of safety. In general, therefore, no changes in shielding of existing sets will be required. However, the recommended limit will insure that future television receivers, operating at higher voltages, will not contribute significantly to the population gonad dose.

(2) In its statement of 8 January 1957 and 15 April 1958, the NCRP specified a permissible dose of $10(N - 18)$ rems to the skin of the whole body for radiation workers. It has subsequently been realized that the use of the age-proration formula for the skin of the whole body is not practical under some conditions. Therefore, the Committee will discontinue this statement and in its place make the following recommendations:

Skin of the whole body: The maximum permissible dose to the skin of the whole body shall not exceed 30 rems per year and the dose in any 13 consecutive weeks shall not exceed 10 rems.

This statement is essentially the same as in the 1959 ICRP report except that the ICRP specified a 13-week limitation of 8 rems. It is felt that this is unnecessarily low if the annual limit is 30 rems.

(3) The above changes will be incorporated in the appropriate reports of the NCRP as the case arises.

Letter from ROLF SIEVERT

It is my unhappy duty to inform you that the Chairman Emeritus of the ICRP, Sir Ernest Rock Carling, died suddenly on July 15.

Although Sir Ernest had reached an age when most people have finished their work, he was still very active and his advice was of great value for the ICRP, not least in administrative matters. Sir Ernest took the chairmanship of the ICRP in a very critical period and his fine personality and administrative capability gave the Commission a fortunate start after World War II. The Members of the ICRP will always remember our Chairman Emeritus as the outstanding leader of the ICRP and look back upon the period of Sir Ernest's chairmanship with gratefulness and admiration.

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* "X-ray emission from television sets" CARL B. BRAESTRUP and RICHARD T. MOONEY, *Science*, **130,** 1071 (1959).

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INTERNAL RADIATION DOSE MEASUREMENTS IN LIVE EXPERIMENTAL ANIMALS—I*

R. L. HAYES, M. M. NOLD,† C. L. COMAR† and H. KAKEHI‡

(*Received* 10 *February* 1960)

Abstract—A study has been made of the use of Schulman–Etzel small-volume dosimeters in the measurement of tissue dose in live animals administered pure β-emitting radioisotopes. The possible extension of this method to similar measurements with mixed β–γ and pure γ-emitters is discussed. The technique developed for the measurement of dose to the intestinal mucosa of animals given oral doses of pure β-emitters is described. The technique involves surgical implantation of dosimeters in the walls of the intestinal tract followed by recovery of the dosimeters after dosing. The application of certain correction factors to the measured response of the recovered dosimeters makes it possible to calculate the dose received by the surface of the intestinal mucosa.

INTRODUCTION

LITTLE attention has been given to the direct measurement of tissue dose in the living animal because of the experimental difficulties. The value of such measurements would lie in providing a firmer basis for correlations between the dosage received at critical volumes within the tissues and the effects of external and internal radiation on animals and man. This type of data should be of some value, especially in the establishment of maximum permissible levels for internally deposited radioactive substances (see National Bureau of Standards Handbook 69 for recent recommendations[1]).

Direct measurements in animals, assuming that they can be extrapolated to man, would provide a means of determining the limits of variability from average behavior either as the result of normal biological variations or as the result of experimental alteration of biological patterns. Further, if the detector is reasonably small, variations in dose produced by non-uniform distributions could be measured. The validity of assuming average behavior in estimating a permissible level of activity and the factors contributing to variations in dose could thus be studied.

The first paper of this sequence describes a technique that has been used for direct dose estimations in experimental animals. The results of these studies, with yttrium-90 in dogs and goats, are presented in Part II.

DETECTOR SYSTEM

The measurement procedure involves the use of Schulman–Etzel small-volume dosimeters[2,3] surgically implanted in the sites to be studied. This technique is well suited to measurements in live animals, since (1) the size of the dosimeter is such that it can often be regarded as a point detector; (2) the dosimeters are not injurious to tissue and hence may be left implanted for long periods of time; and (3) their sensitivity to radiation is such that only moderate doses of radiation are required for the measurements; variations produced by radiobiological disturbances may thus be avoided.

The Schulman–Etzel dosimeters are made from a special silver phosphate glass.[4] They are produced in the form of cylindrical rods 1 mm in diameter and 6 mm long, and are now commercially available from the Bausch and Lomb Optical Company. The principle of measurement is based on the production of permanent luminescence centers as the result of the action of ionizing radiation. On exposure to

* From the Medical Division, Oak Ridge Institute of Nuclear Studies, Oak Ridge, Tennessee, under contract with the United States Atomic Energy Commission and special contract with Armed Forces Special Weapons Project (SW-PEF-5/927).

† Laboratory of Radiation Biology, Cornell University, Ithaca, New York.

‡ Department of Radiology, Chiba University Hospital, Chiba, Japan.

near ultraviolet light, the needles emit an orange luminescence with an intensity proportional to the radiation received by the dosimeter. A special fluorimeter for measuring this luminescence has been designed by the Naval Research Laboratory. An instrument of greater sensitivity known as a "Microdosimeter Reader" has recently been developed by the Bausch and Lomb Optical Company. Both instruments use photomultipliers as luminescent light detectors.

The luminescence response of this dosimeter system is independent of incident photon energy above 250 keV. Below this energy the response increases, reaching a peak at approximately 70 keV, at which point the response of the detector is about twenty times as great as in the energy-independent region.[4]* This characteristic obviously limits the usefulness of the Schulman–Etzel dosimeter; however, the shortcoming may be partially overcome by the use of certain correction techniques.

The luminescence centers produced by ionizing radiation are stable over extended periods of time, and hence the dosimeters are, in effect, integrating detectors. This property is obviously a valuable one for measurements in highly variable biological systems where uniformity is the exception rather than the rule.

The rods exhibit a "predose" luminescence or background which in turn affects the lower limits of detection. This limit necessarily varies with the energy of the radiation being detected. For cobalt-60 radiation the Naval Research Laboratory reader gives a net reading equal to the background reading with a dose of approximately 40 rads.

MEASUREMENTS WITH PURE β–EMITTERS

Implantation directly in active tissue

The Schulman–Etzel small-volume dosimeter is readily adapted to the determination of β-dose. In such measurements there will be a variation in β-flux through the internal volume of the dosimeter and the dosimeter cannot be considered a point detector as is generally possible with X- or γ-radiation. For this reason the response of the dosimeter system per unit of dose received must be determined with the particular β-emitter being studied. This is conveniently done by exposing the dosimeter for a known time in an aqueous solution containing a known concentration of the β-emitter in question. The dosimeter is positioned so that the thickness of the surrounding solution exceeds the maximum range of the β-particles involved. At all points in the calibrating solution that correspond to this positioning, the rate of energy production will be equal to energy absorption. The dose in rads will then be given by:

$$D = 74\,\bar{E}TC\,(1 - e^{-0.693\,t/T}) \qquad (1)$$

where D = dose (rads);
$\bar{E}$ = average β-energy (MeV);
C = initial concentration of β-emitter (μc/g);
T = half-life (days);
t = time exposed (days).

The measured luminescence of the dosimeter will thus represent its response for a condition of uniform emitter distribution in a system approximating tissue. The effect of Bremsstrahlung is disregarded. Fig. 1 shows a study of the luminescence response of the Schulman–Etzel dosimeter system to yttrium-90 β-radiation in aqueous solution. The rods were suspended by fine nylon threads so that there was minimum contact between the surface of the dosimeter and the supporting thread. Once the luminescence response for a particular β-emitter has been determined, a secondary γ-standardization can be used as a matter of convenience.

In actual measurements in tissue the concentration of radioactivity need not be constant, but may change with time in any pattern so long as the concentration change is a uniform one within the prescribed β-range distance from the dosimeter. In practice, only small errors

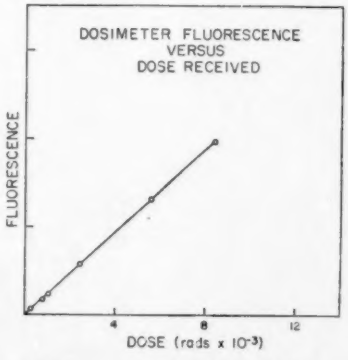

FIG. 1. Fluorescence of the Schulman–Etzel dosimeter as a function of dose.

* Experimentally determined values.

are involved by breakdown of this condition at the outer limits of the β-range, since the response contribution at these distances is only slight.

Implantation adjacent to active tissue

If the dosimeter is implanted in tissue that does not contain radioactive material itself but is adjacent to such tissue, special correction procedures can be employed. An example of this condition is the measurement of β-dose to the surface of the intestinal mucosa by implantation of the dosimeter in the wall of the intestine. Correction factors for the effective geometry and the absorption of β-particles by the intervening tissue on the luminescence response of the implanted dosimeter must be used. If it is assumed that the interface between the active and inactive tissue is a plane and that the implanted dosimeter lies parallel to this plane in the inactive tissue, the luminescence of the dosimeter may be related to the dose at the interface by using a phantom technique. The luminescence responses of dosimeters separated by various thicknesses of tissue-equivalent plastic from the surface of an infinitely thick solution of the β-emitter in question are compared with the response of a dosimeter suspended directly in the solution. If the distance of the suspended dosimeter from the containing walls of the phantom exceeds the β-range, for reasons of symmetry, the response of the suspended dosimeter will be twice that of a dosimeter placed at the interface between the solution and the phantom wall. In this way it is possible to correct the measured response of recovered implanted dosimeters to the response they would have had at the surface of the mucosa. Fig. 2 shows such correction curves obtained for yttrium-90 and strontium-89 by this phantom technique.

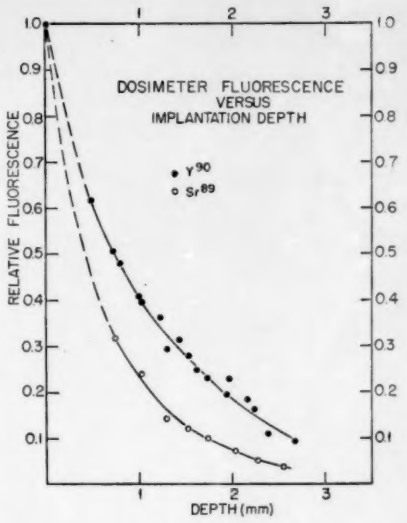

FIG. 2. Effect of tissue implantation depth on fluorescence response of Schulman–Etzel dosimeter rods to Y^{90} and Sr^{89} β-radiation.

MEASUREMENTS WITH β–γ EMITTERS

Where β-dose is to be measured in the presence of associated γ-rays it should be possible to determine the luminescence contribution of a γ-component by using dosimeters shielded against the β-particles in question. Subtraction of this contribution from the luminescence of the unshielded dosimeter would then yield the luminescence due to the β-dose. The β-dose could thus be determined by paired implantations of shielded and unshielded dosimeters in experimental animals.

The use of the Schulman–Etzel small-volume dosimeter in γ-dose measurements in experimental animals involving either mixed β- and γ- or pure γ-emitters is complicated by the previously mentioned energy dependence of the dosimeter system. The energy-dependence problem can be overcome to a great extent by encasing the dosimeter in a shield of high atomic number to absorb preferentially most of the low-energy component and in effect decrease the energy dependence of the dosimeter.[4,5]

Dose measurements involving high-energy γ-radiation are less subject to energy-dependence error. Schulman–Etzel small-volume dosimeters have been used routinely at this installation for skin-dose measurements on patients receiving total-body cobalt-60 treatments.

SPECIAL CONSIDERATIONS

Fading

A study was made of the luminescence stability of exposed dosimeters in an initial investigation

to confirm the reported characteristics of Schulman–Etzel small-volume dosimeters before their experimental use. Contrary to expectations, the dosed dosimeters showed a slow apparent loss of luminescence on standing.

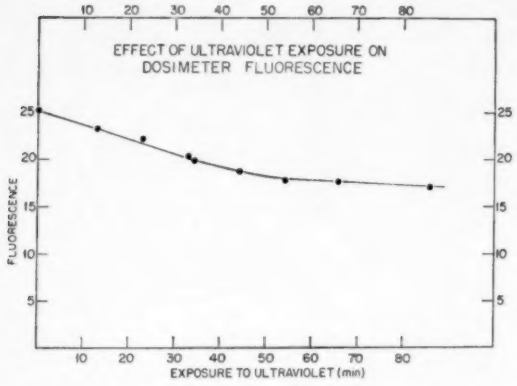

FIG. 3. Fluorescence of an irradiated Schulman–Etzel dosimeter as a function of time of exposure to ultraviolet light in the NRL dosimeter reader.

Further study revealed that this fading was brought about by the cumulative effect of the ultraviolet-light exposures resulting from the luminescence measurements rather than through the instability of luminescence centers on standing at room temperature. Fig. 3 shows the effect of prolonged ultraviolet-light exposure in the dosimeter reader on the luminescence response of a typical dosed dosimeter. Temperature effects on luminescence were excluded by removing the dosimeter from the reader and allowing the dosimeter to come to room temperature before each redetermination of luminescence was made. These ultraviolet-light effects were observed with the dosimeter reader designed by the Naval Research Laboratory. In actual practice, if the ultraviolet-light exposure is minimized through rapid readings, the loss of luminescence is negligible with this instrument.

One of the writers (M. N.) has observed a similar effect, but to a much lesser degree, when using the new Bausch and Lomb Microdosimeter Reader. The effect of ultraviolet light with this instrument can be disregarded.

Standardization

Since the output of a photomultiplier is highly dependent on the applied voltage, it is necessary to use some standard method for setting the voltage to obtain a reproducible output. Irradiated dosimeter rods could not be used with the Naval Research Laboratory reader because of their tendency to fade with prolonged exposure to ultraviolet light. Consequently, an instrument reference standard approximating the dimensions of the Schulman–Etzel dosimeter rod was made from Corning "fluorescent canary" glass (filter no. 3750). This glass emits a luminescence similar to that of dosed dosimeters. The use of a special manganese glass for standardization has also been reported.[6]

The luminescence of this reference standard as a function of time in the instrument for various prior warm-up times is shown in Fig. 4. As Fig. 4 indicates, in standardizing the Naval Research Laboratory reader, the voltage should be adjusted rapidly after introduction of the standard in the reader. If more accuracy and assurance in the instrument standardization are required, the voltage can be adjusted after the reference has been inserted for approximately

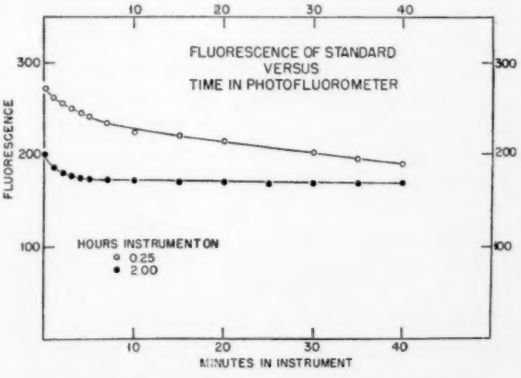

FIG. 4. Behavior of uranium glass standard in NRL dosimeter reader.

10 min when the instrument has had a prior lamp warm-up of approximately 2 hr. Where readings are to be made on a large number of dosimeters the latter procedure should be adopted, since this will eliminate the frequent

restandardizations that would otherwise be required.

Dosed dosimeters exhibited an "in-instrument" behavior similar to that of the instrument standard. It was impractical from a time standpoint to allow the dosimeters to reach equilibrium luminescence before reading, since multiple readings of each dosimeter were required for precision in measurement. Consequently the readings were taken as rapidly as possible after each insertion of the dosimeter in the instrument. "In-instrument" time for such readings was approximately 15–20 sec.

In a further effort to obtain a consistent behavior from day to day the instrument, instrument standard and dosimeters to be read were kept in a constant-temperature room.

Tissue effects

Studies designed to check the constancy of the "predose" background luminescence of Schulman–Etzel dosimeters before and after implantation in animals showed that rods implanted for periods of up to 8 weeks increased in background luminescence by as much as 50 to 100 per cent. This increase in background luminescence was associated with a decrease in the surface luster of the needles. Apparently contact with tissue or tissue fluids results in surface etching. This would be of little significance where high radiation doses are involved. On the other hand, considerable error could be introduced in dose measurement at low levels of radiation exposure.

In an effort to overcome this difficulty, a technique was devised for artificially etching the surfaces of dosimeters before their implantation. When such pretreated dosimeters were implanted in experimental animals and harvested within 6 to 8 weeks after implantation, no further additive tissue effect was observed. No significant difference in dose response between pretreated and untreated dosimeters was observed.

The treatment consisted of soaking the dosimeters for 24 hr in a 0.5% solution of "Haemosol", a commercially available detergent for cleaning laboratory glassware. Fig. 5 shows the results of this type of pretreatment on the background response or "predose" luminescence of a typical dosimeter.

In a further study of tissue effect under conditions of long-term implantation, dosimeters were kept implanted for periods of from 7 to 7½ months before harvesting. Even though these rods had been given the pre-implantation treatment, there was an additional twofold to fourfold increase in background response. Deep erosion of the surface of these dosimeters was clearly visible under low-power magnification. It appears from these results that the use of these dosimeters for tissue-dose studies is limited to relatively short implantation times (up to 6 or 8 weeks). Their use for long-term studies involving low doses is not feasible unless procedures are developed to compensate or prevent the increased surface erosion that is encountered.

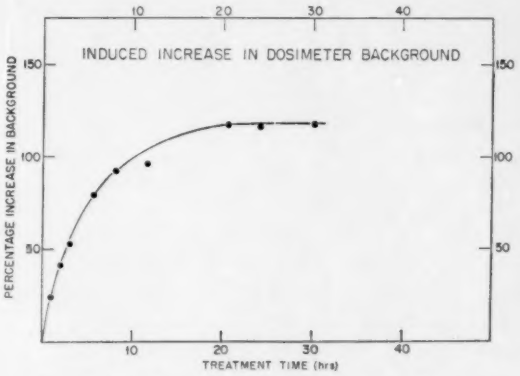

FIG. 5. Typical induced increase in dosimeter background by "pretreatment" with surface etching solution.

MEASUREMENT OF DOSE TO THE INTESTINAL MUCOSA

The Schulman–Etzel small-volume dosimeter has been used to make measurements, in dogs and goats, of the dose received by the intestinal mucosa as the result of oral ingestion of pure β-emitters. The results of such measurements using yttrium-90 are reported in Part II of this paper. The technique is generally applicable to high-energy β-emitters.

The technique involves the surgical implantation of Schulman–Etzel small-volume dosimeters in the walls of the intestinal tract of

experimental animals. After recovery from surgery, a known amount of activity is administered to the animal and the dosimeters are subsequently recovered. The dose received at the various implantation sites is then determined by measuring the luminescence response of the recovered dosimeters.

All dosimeters to be used were inspected under a low-power microscope and any rods with obvious surface flaws were discarded. Dosimeters to be implanted were allowed to soak for 24 hr in a 0.5% solution of "Haemosol", after which they were rinsed repeatedly in distilled water and finally in acetone. Recovered implanted dosimeters were washed thoroughly in detergent solution (Tide), rinsed repeatedly in distilled water, and finally in acetone.

The Naval Research Laboratory dosimeter reader was standardized as follows: The instrument was turned on (including the ultraviolet lamp) and allowed to warm up for 2 hr or more. The standard uranium-glass reference mentioned in the foregoing discussion on standardization was then inserted. Ten minutes after insertion the response of the standard was brought to the chosen standard response by adjustment of the voltage on the photomultiplier tube. The "predose" luminescence of each dosimeter was taken as the average of a series of eight response readings. Four readings were taken; the dosimeter was rotated approximately 90 degrees about its vertical axis between these readings. The dosimeter was then inverted in the holder and four similar readings were taken. All readings were taken rapidly (within 15–20 sec); the dosimeter was removed from the instrument immediately after the maximum response was noted. Readings on recovered implanted dosimeters were made in the same way.

Implantation of the dosimeters in the intestinal tracts of experimental animals was carried out as follows: Each animal was fasted for 24 hr before surgery, anesthetized, and the abdomen was then prepared for surgery. A midline incision was made through the abdominal wall and that portion of the gastrointestinal tract to be implanted was withdrawn. The overlying omentum was freed from the area and the gut section was withheld exteriorly by insertion of curved forceps through the adjacent mesentery. A 1-cm incision was made in the gut wall transverse to the long axis of the tract and to the depth of the submucosa. Two anchoring sutures (6-0 noncapillary silk) were then placed through the submucosa about 3-4 mm apart and centrally located in the incision. The dosimeter was placed on the inserted sutures, which were then firmly tied around it. The wound in the gut wall was closed over the dosimeter with a single row of inverting sutures. A small amount of India ink was then placed in the incision to aid later recovery of the rod. The withdrawn gut section was replaced in the abdomen and the abdominal incision was closed. All surgery was performed under aseptic conditions with a minimum of handling and trauma. Fig. 6 illustrates the steps involved in the implantation procedure. The animal was fed only milk the day after the operation, and thereafter its usual ration of food. Normally from 10 days to 2 weeks was allowed for recovery before administration of the radioactivity. Known doses of radioactivity were normally administered mixed with the animals' food.

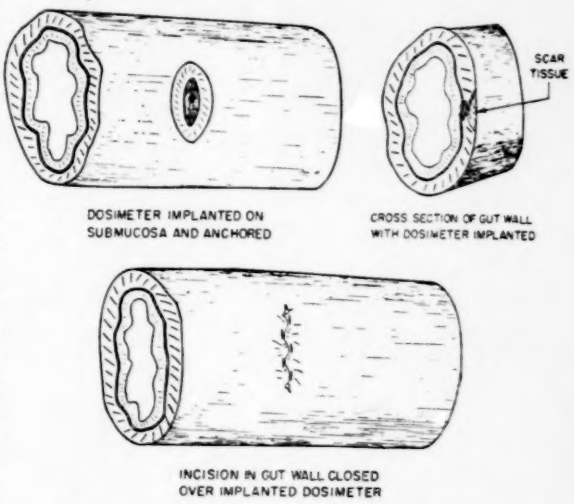
SCAR TISSUE
DOSIMETER IMPLANTED ON SUBMUCOSA AND ANCHORED
CROSS SECTION OF GUT WALL WITH DOSIMETER IMPLANTED
INCISION IN GUT WALL CLOSED OVER IMPLANTED DOSIMETER

FIG. 6. Surgical implantation of Schulman–Etzel dosimeter rod in wall of the intestinal tract.

The thickness of tissue between the rod surface and the mucosa surface was carefully

determined for each implanted needle at the time of recovery. The use of a seven-power magnifying monocular eyepiece made it possible to measure the tissue thickness to 0.1 mm.

Each recovered dosimeter was cleaned and its luminescence response was determined in the manner already described. In calculating the dose to the mucosa, the net luminescence (luminescence response minus "predose" response) was first corrected for implantation depth by using the correction curve shown in Fig. 2. The actual mucosa dose was then calculated by the use of an instrument calibration factor obtained by the technique described in the preceding discussion. In formula form the dose is given by:

$$D = \frac{r_2 - r_1}{R \times f} \qquad (2)$$

where D = dose to the surface of the gastrointestinal mucosa (rads);
r_1 = "predose" luminescence of dosimeter;
r_2 = luminescence of recovered implanted dosimeter;
R = dosimeter luminescence per rad in reader;
f = relative luminescence for dosimeter implantation depth as measured at time of recovery (Fig. 2).

This method for relating the luminescence of recovered implanted dosimeters to the dose received by the intestinal mucosa involves the following assumptions: (1) No significant effect on the luminescence of implanted dosimeters is produced by the presence of Bremsstrahlung or by the penetration of β-particles from adjacent parts of the intestinal tract. (2) The mucosal surface is assumed to be smooth, and any geometry effects due to the curvature of the intestinal walls are neglected. (3) The inside diameter of the intestinal tract at the site of implantation is assumed to exceed the maximum β-particle range for the emitter involved. (4) A further inherent assumption is that during the passage of any experimental dose there is no change in the thickness of tissue between the implanted dosimeter and the surface of the mucosa as the result of swelling caused by radiation damage.

Acknowledgments—The writers wish to acknowledge the generous co-operation of Dr. J. H. SCHULMAN of the Naval Research Laboratory.

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INTERNAL RADIATION DOSE MEASUREMENTS IN LIVE EXPERIMENTAL ANIMALS—II*

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Abstract—Silver phosphate glass dosimeter rods were implanted in various portions of the digestive tract and the radiation dose was measured after ingestion of a known amount of yttrium-90. It was found that a state of diarrhea reduced the average radiation dose by a factor of from 2 to 4. In the constipated animal the dose was increased by a factor of from 3 to 7.

Investigation was made to determine the role of various processes governing the radiation dose delivered to gastrointestinal mucosa. The total dose to a particular site along the intestinal tract was obtained by determination of the time integral of the radioactive concentration. Serial sacrifices were made at specific times after administration of the radioactivity. Calculations in this manner agreed exceptionally well with the doses that were measured by the glass dosimeter method.

It is estimated that 4 μc and 17 μc of yttrium-90 for the dog and goat, respectively, will deliver a 300-mrad dose to the critical organ, the lower large intestine. The twelvefold average difference in dose between the diarrhea and constipation groups of dogs emphasizes the importance of the physical state of bowel passages upon the dose delivered to the critical organ.

1. INTRODUCTION

THE response of the gastrointestinal tract to either external radiation or ingested radioactive materials is of considerable importance. It is generally recognized that one of the most immediate effects of radiation exposure is manifested by the gastrointestinal tract. This effect results from exposure to external sources of ionizing radiation as well as from ingested radioactivities. In the past it has been difficult or impossible to obtain direct physical measurement of the dose received by the intestinal mucosa. The present maximum permissible concentration (MPC) values for those isotopes that have the large intestine as their critical organ are, to a large extent, calculated estimations. It seemed highly desirable to check these MPC values for at least one radionuclide, as presently stated, with data from biological experiments. Under varying conditions, such as diarrhea and constipation, the dose to the critical organ from a defined amount of radioactivity may be greatly altered. It was also of interest to compare results from ruminants (goats) with those from a single-stomached species (dogs).

This paper is primarily concerned with the estimation of radiation dosage to the gastrointestinal mucosa from the ingestion of poorly absorbed radioactive materials. Results are reported for direct dose measurements following administration of yttrium-90 to the dog and the goat. There is no significant absorption of yttrium from the intestinal tract.[1] The technique of using the silver phosphate glass-rod dosimeter system in direct dose measurements in the live animal has been described in Part I of this paper. In addition, data are presented on an alternate experimental method of estimating the dose to the various parts of the gastrointestinal tract.

II. DIRECT DOSIMETRIC MEASUREMENTS

A. *Sites of implantations*

In preliminary experiments, as many as six

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dosimeters were implanted in various parts of the tract of each animal: stomach, duodenum, ileum, cecum, mid-colon, and terminal colon (from 4 to 6 in. from the anus). Also, an occasional dosimeter was placed in abdominal muscle fascia to check any change in background reading of the needle that might have been caused by tissue reaction. The first studies using yttrium-90 showed that the large colon always received the highest dose, and, therefore, all subsequent dogs were implanted in the mid-colon and terminal colon only. Two dosimeters were placed in each of these locations to compare readings and to insure against loss of data through accidental loss or breakage.

The surgical techniques with the goat were basically similar to those described for the dog. It was advantageous to withhold all food from the goat for 3 days before surgery and to withhold all water for the previous 24 hr.

B. *Isotope administration and handling*

After the dog had recovered from surgery and exhibited normal appetite and bowel movements, the radioactive material was administered. The yttrium-90, as the chloride in solution, was mixed with a small amount of dry dog food; this mixture was then thoroughly mixed with the usual ration and fed to the dog. The radioisotope preparations were routinely purchased from and standardized by Abbott Laboratories, Oak Ridge, Tennessee. Five milligrams of stable yttrium was added to each dose as a carrier. Studies with stable yttrium showed that no appetite or laxation effects were caused by the stable element content of the dose.

Monitoring of the feces of physiologically normal dogs showed that practically all the yttrium-90 was eliminated within 1 to 7 days after administration. The animal was routinely killed on the seventh day, the dosimeters were located, and a careful measurement was made of the implantation depth (Table 1). For this purpose, a seven-power magnifying monocular eyepiece was used. This eyepiece contained a graduated micrometer that permitted thickness measurements to 0.1 mm accuracy. It was necessary to avoid scratching the dosimeter surface during handling since this would interfere with measurement of the fluorescence.

Table 1. *Range of implantation depths* (mm)

No. of dogs	Mid-colon	Terminal colon
36	1.26 ± 0.03*	1.31 ± 0.04*

* Standard error of mean

In most of the goats used in this study, the yttrium-90 (as the chloride solution) was injected into the contents of the rumen through the left flank. A 3-in. 18-gauge needle attached to a 10-ml hypodermic syringe was used, and the syringe was routinely flushed three times with water after the injection of the dose.

Some of the goats were given radioactivity orally. In these, the yttrium-90 was placed on Whatman accelerator paper and vacuum dried. The paper was then placed in a gelatin capsule and administered by balling gun. Comparisons of concentration–time values by these two methods of administration showed that there were only small differences in the doses delivered.

C. *Tissue swelling*

Because of sensitivity of the measurement to implantation depth, it was necessary to determine whether the radiation dose would cause significant swelling. The time relationships between the occurrence of the swelling and the deliverance of the dose also had to be considered.

Each of six dogs (from 20 to 40 lb) was given 25 mc of yttrium-90 in a single feeding. They were killed at 24-hr intervals after dosing and the mucosal depths of the mid-colon and terminal colon were measured. Two control animals were also used to give the initial values. The results are shown in Fig. 1. It is noted that there was an increase in tissue thickness with the peak occurring at from 2 to 3 days after feeding the radioactivity and a return to normal after the fifth day. Since the largest amount of radioactivity usually is passed in the feces within 24 hr, it appears that the major portion of radiation is received by the tissue at a time when the mucosa is only slightly swollen. Accordingly, no attempt was made to correct for this factor.

D. *Production of diarrhea and constipation in dogs*

The normal dog diet consisted of commercial

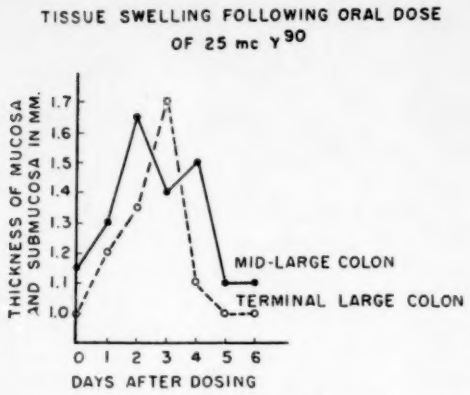

FIG. 1. Thickness of mucosa and submucosa after ingestion of a single dose of yttrium-90.

canned food with fresh water *ad lib*. The animal was placed on this ration several days before isotope administration to permit gastrointestinal regulation. All dogs, when under study of normal conditions, were passing formed stools daily.

In the experimental animals a state of diarrhea (nonformed stools) was attained by the addition of magnesium oxide to the ration in sufficient quantity (1–2 g daily) to produce nonformed bowel movements.

Considerable difficulty was experienced in obtaining a controlled state of constipation. The use of paregoric, metropine, pectin, hexamethonium chloride, and limited water intake resulted in uncontrolled bowel movements.

Best results were obtained when a bulk-free diet was fed. This diet was composed of dextrose 3 g, dextrin 3 g, sucrose 3 g, lard 3.5 g, casein 3 g, salt 0.3 g, and water in a sufficient quantity to give the consistency of a thick gravy; this amount was fed daily per kilogram of body weight. Even with this nonbulk diet considerable variation (from two to five) was noted in the number of stools passed during the week following isotope administration. An occasional "constipation" dog would develop diarrhea 3 or 4 days after ingestion of radioactivity. Since the radiation delivered by the routine dose of 25 mc of yttrium-90 could have been the cause of this diarrhea, some of the later animals were given only a 5-mc dose.

E. *Dose to various parts of the tract*

Table 2 shows the doses received by various parts of the gastrointestinal tract of medium-sized dogs after ingestion of 25 mc of yttrium-90.

The colon received the highest dose, from 800 to 2100 rads; this was from two to four times higher than for the cecum, and about ten times higher than for the stomach and duodenum, which received the lowest dose. Since the mid-colon and terminal colon received the highest dose by a considerable factor, only these two regions were studied on a routine basis in the remaining experiments.

1. *Single vs. multiple dose*. A practical aspect is concerned with the dose received under conditions of daily ingestion. As a check on the additivity of dose, some studies were done in which 25 mc of yttrium-90 were given at a rate of 5 mc/day over a period of 5 days. The dosimeters were then recovered 7 days after the

Table 2. Radiation dose in normal dogs after ingestion of* 25 mc *of yttrium-90*
(Values in rads)

Dog. no.	456	466	467	550	472	563	Mean‡
Type dosage†	Single	Single	Single	Single	Multiple	Multiple	
Stomach	300	315	215	—	115	90	205 ± 46
Duodenum	250	110	50	—	50	25	95 ± 41
Ileum	570	630	550	—	505	100	470 ± 95
Cecum	500	665	—	—	265	250	420 ± 99
Mid-colon	910	1580	1080	805	1460	—	1165 ± 152
Terminal colon	—	2080	1990	1160	2060	2040	1865 ± 178

* From 20 to 40 lb body weight; canned dog food diet; daily formed stools.
† Single ingestion of 25 mc yttrium-90 or five daily feedings each of 5 mc of yttrium-90.
‡ Mean ± standard error of mean.

Table 3. Effect of body size on dose after ingestion of 25 mc of yttrium-90*
(Values in rads)

	No. of dogs	Mid-colon**	Terminal colon**
Large† (>55 lb)	2	1020 ± 416	1015 ± 446
Medium‡ (20–40 lb)	5	1165 ± 152	1865 ± 178
Small§ (<16 lb)	4	2415 ± 466	2430 ± 346

* Canned dog food diet; daily formed stools.
† Fed 2 cans of dog food daily.
‡ Fed 1 can of dog food daily.
§ Fed ½ can of dog food daily.
** Mean ± standard error of mean.

last ingestion of radioactivity. As shown in Table 2, approximately the same results were obtained whether the activity was administered as a single dose or as a daily dose over the 5-day period.

2. *Effect of body size.* The results in Table 3 show that the larger the dogs, the smaller the dose received by the large colon. The medium and large dogs had about 60 per cent and 40 per cent, respectively, of the dose received by the small animals.

Probably the body size–dose relationships are due, to a large extent, to the concentration of the radioisotope in the food mixture at the time of administration. The dose (25 mc) was routinely mixed with less food (half a can) for the small animals and more food for the medium size and large dogs (from one to two cans).

3. *Effect of diarrhea and constipation.* Since the irradiation dose to the gastrointestinal tract is directly proportional to the time during which the exposure occurs and the concentration of radioactivity, it seemed that a state of diarrhea or constipation should have a noticeable effect.

The normal dogs (Table 4) were placed on the basal diet of commercial canned dog food several days before isotope administration to permit bowel regulation and the daily passage of formed stools. Likewise, the diarrhea and constipation animals were placed on their respective diets several days before dosing for attainment of the desired states of bowel movements.

It is clear from Table 4 that a condition of diarrhea decreased the average radiation dose to the colon by a factor of from 2 to 4. The response was the same whether the diarrhea was produced on the normal diet or on the nonbulk diet. Conversely, a condition of constipation increased the average radiation dose to the colon by a factor of from 3 to 7. Note that in one constipated dog, a dose of almost 15,000 rads was received as compared with the average of about 1200 rads for the normal animals. It should be emphasized that the degrees of constipation in this group varied greatly. There was good correlation between the dose received and the timing of the bowel movements following isotope administration.

Nevertheless, the doses obtained in the constipation group of dogs are probably considerably lower than the true delivered doses because of tissue swelling. Reference is made to Fig. 1, which indicates that the maximum amount of swelling occurs from 2 to 3 days after administration of 25 mc of yttrium-90 to dogs with normal bowel movements. Under conditions of constipation the increased radiation dose to the mucosa probably would result in a greater amount and a lengthened period of swelling. The correction for this increased distance between the surface of the mucosa and the implanted dosimeter during a portion of the exposure period could not be included in the tissue depth measurement at the time of dosimeter harvest. Therefore, the actual dose delivered to the mucosa of the large intestine would be expected to be somewhat higher than indicated in Table 4. An accurate estimate of this increase is impossible with the present data.

F. *Studies with the goat*

Goats were used to obtain data on another species with different feeding and stomach characteristics. The procedures were similar to those described for the dog. The goats averaged from 70 to 80 lb body weight and the amount of yttrium-90 administered was 100 mc. For purposes of comparison with the data on dogs, however, the results as seen in Table 5 are expressed in terms of 25 mc ingested dose.

It is noted from Table 5 that, of the various

Table 4. Effect of diarrhea and constipation on radiation dose in medium-sized dogs after ingestion of 25 mc of yttrium-90
(Values in rads)

	No. of dogs	Dog no.	Mid-colon	Terminal colon	Pathology
Normal*	5		1165 ± 150	1865 ± 180	—
Diarrhea,	6	547	260	250	—
normal diet†		548	300	355	—
		553	1435	1210	—
		554	520	265	—
		563	685	790	—
		564	515	355	—
			Avg. 620	540	
Diarrhea,	5	568	730	580	—
nonbulk diet‡		569	465	565	—
		571	580	495	—
		570§§	2415	2215	—
		577***	2807	3975	+
	(excluding 570 and 577)		Avg. 590	545	
Constipation,	8	555	14690	11,540	+
nonbulk diet§		556	7720	11,590	—**
		558	6140	4830	
		559	12450	6515	+
		565	6615	—	—**
		561	4025	5065	+††
		562	3320	2490	+††
		566	2800	2240	—**‡‡
			Avg. 7220	6325	

* Canned dog food.
† Canned dog food plus MgO.
‡ Dextrose, dextrin, sucrose, casein, lard, salt plus MgO.
§ As above but without MgO.
** These dogs received only 5 mc of yttrium-90 so that actual radiation dose was one-fifth that listed.
†† These dogs had daily bowel movements after dosing.
‡‡ Developed diarrhea after first day after dosing.
§§ Refused all food on fifth day after dosing and appetite had not returned to normal when killed on the fourteenth day.
*** No explanation for the high dose.

Table 5. Radiation dose to various parts of the gastrointestinal tract of goats after oral dose of yttrium-90
(Corrected to 25 mc dose; values in rads)

Goat no.	3	5	6	9	12	13	14	15	16	Avg.
Rumen	—	99	—	94	—	31	99	52	194	95
Reticulum	110	57	—	103	—	—	—	—	—	90
Abomasum	35	42	—	—	—	—	—	—	—	39
Small intestine	—	—	—	—	—	72	113	107	97	97
Cecum	—	—	354	—	125	167	659	292	376	329
Large intestine	—	—	—	—	—	112	—	239	—	176
Terminal colon	—	—	774	—	—	264	519	199	418	435

sections, the cecum and colon received the highest radiation dose, but not by so large a factor as for the dogs. It is of particular interest to note that for a given dose of radioactivity the terminal colon of the goat received only about one-fourth the radiation dose received by that tissue in the medium-sized dog.

G. *Pathology*

Routine pathologic examination was made on the colon tissues adjacent to the site of dosimeter implantations. The results are summarized by groups of animals.

1. *Normal dogs.* Killed daily after ingestion of 25 mc of yttrium-90.

(a) Controls. No significant pathology noted.

(b) At 1 day. A few nuclear fragments noted at base of crypts.

(c) At 2 days. Neutrophilic leucocytes noted in mucosa; tips of villi absent; some nuclear fragments in crypts.

(d) At 3 days. Tips of villi sloughed.

(e) At 4 days. Eosinophils noted in base of crypts; interstitial neutrophils; nuclear debris in glands; definite inflammation present.

(f) At 5 days. Neutrophilic leucocytes in stroma; nuclear debris in lumen; base of crypts nearly intact.

(g) At 6 days. Essentially normal.

2. *Normal and diarrhea dogs*; 25 mc *of yttrium*-90 (*from* 400 *to* 2700 rads). Killed at seven days. An occasional animal showed neutrophilic leucocytes in stroma; one animal out of thirteen demonstrated a very flattened epithelium. Otherwise the tissues were unremarkable.

3. *Constipation dogs*; 25 mc *of yttrium*-90 *dose* (*from* 6000 *to* 15,000 rads). Killed at seven days. Blood vessels dilated; loss of epithelium; stroma contained many neutrophilic leucocytes; telangiectasis noted in some of the higher dosed animals. Grossly, the mucosa showed hemorrhagic inflammation.

Some of the dogs in this group received only 5 mc of yttrium-90, but all results were calculated on a 25 mc basis for comparison. In this subgroup the actual dose did not exceed 2700 rads, and no remarkable lesions were noted.

No pathologic changes were observed in the tissues from the goats. In this connection it must be remembered that the actual radiation doses received were four times those listed in Table 5.

III. RELATIONSHIPS OF DOSAGE AND INGESTA DYNAMICS

A. *General*

The next logical step was to determine the role of the various processes governing the radiation dose received by the gastrointestinal tract. The factors contributing to the dose relationships of the various parts of the tract are:

1. *Food intake.* This is very important since, for a given amount of ingested β-emitting activity, the concentration of radioactivity in the ingesta is inversely related to the volume of the food ingested. This is especially significant for the dose delivered to the upper part of the intestinal tract. Variations in dose might be expected in the lower parts of the digestive tract because of food composition as well as species differences.

2. *Fluid dilution.* The dilution of the ingesta both by water intake and internal digestive secretions results in a lowered concentration of radioactivity in the ingesta. This leads to decreased dose since the dose is directly proportional to the concentration of radioactivity. Dilution effects would be more noticeable in the upper parts of the tract.

3. *Hold-up time.* The relative length of time that the intestinal contents are in contact with a particular section of the gastrointestinal mucosa will directly govern the dose received. If all other factors including decay are neglected, the organ of greatest hold-up time will receive the largest dose. In the dog it is known that the stomach empties rather rapidly and that ingesta remain longer in the lower portion of the tract than elsewhere. In the normal goat, the rumen would be expected to be the rate-controlling portion of the tract for elimination. Neglecting all other factors including decay, one would thus expect those sections of the tract posterior to the rumen, which always contain ingesta (i.e. cecum and the large intestine), to receive no greater dose than that of the rumen.

4. *Physiological concentration of radioactivity.* The concentration of radioactivity occurs by absorption of nutrient materials from the intestinal

tract and by dehydration of the intestinal contents, particularly in the lower part of the digestive tract. Concentration by these two paths would increase the dose to the lower parts of the intestinal tract relative to the stomach for both the dog and the goat.

5. *Radioactive decay.* The half-life of the ingested radioisotope could contribute heavily to the dose relationships of the various parts of the tract. With a radioisotope having a half-life of several weeks or longer, the decay that would occur during passage time through the gut would be negligible. On the other hand, with a short-lived radioisotope, the bulk of the energy expended by the decay process might occur before the terminal portions of the intestinal tract were reached by the contaminated ingesta. In mixed fission products, only those that had been recently formed (e.g. containing a large percentage of short-lived radioactivities) would cause an increased comparative dose to upper parts of the gut.[2]

B. *Experimental plan*

A study was undertaken to allow the determination of doses to the dog and goat in addition to investigating some of the factors in governing radiation dose. The total dose to a particular site along the intestinal tract may be obtained from a knowledge of the variation of the dose rate with time, i.e. by determination of the time integral of concentration. Using the same assumptions as in Part I of this paper the dose D, in rads per millicurie ingested, is given by:

$$D = \frac{Kh}{2} \qquad (1)$$

where K = g rads/μc hr for Y^{90};

h = μc hr/g of exposure per mc ingested.

Calculations based on an average energy of 0.895 MeV[3] for yttrium-90 give a K-value of 1.91 g rads/μc hr.

The yttrium-90 was administered in a manner similar to that used with implanted animals. Serial sacrifice at specific times after administration of the activity permitted determination of the concentration of the radioactivity as the ingesta passed through the digestive tract. Steady-state ingesta concentrations were obtained from another series of animals.

C. *Methods*

The dogs and goats used in this phase of work were handled the same as were those described previously in the direct dosimetric study except that no surgery was performed.

All animals were killed at the given time period with pentobarbital sodium administered intravenously, and collection of the intestinal contents was begun immediately. The dog's tract was sectioned as follows:

(1) Stomach (S)
(2) Small intestine (SI 1/2; first half).
(3) Small intestine (SI 2/2; second half).
(4) Cecum (C).
(5) Large intestine (LI 1/2; first half).
(6) Large intestine (LI 2/2; second half).

The contents were manually stripped out and weighed. After thorough mixing with a spatula, a sample was obtained from each of these sections. Two drops of toluene were added to each sample bottle as a preservative.

The goat's digestive tract was divided into the following sections:

(1) Rumen (Ru).
(2) Reticulum (Ret).
(3) Omasum (O).
(4) Abomasum (Abo).
(5) Small intestine (SI 1/2; first half).
(6) Small intestine (SI 2/2; second half).
(7) Cecum (C).
(8) Large intestine (LI 1/3; first third).
(9) Large intestine (LI 2/3; second third).
(10) Large intestine (LI 3/3; third third).
(11) Terminal colon (TC; last 10–12 in. of tract)

As with the dog, the contents were stripped out and weighed. The ingesta sample was thoroughly mixed before the collection of the sample to be assayed.

A weighed amount of each sample was digested with hot concentrated HNO_3 and diluted to a known volume (50 ml). Ten milliliters of this solution was used for routine counting at infinite sample thickness for yttrium-90 β-rays. All results were corrected for decay to a common counting time and counted with an end-on

G.–M. tube according to commonly accepted laboratory techniques.

These results thus represented the concentrations for an infinite half-life (stable yttrium). Yttrium-90 decay could then be applied to these results when a composite picture of the concentration and decay processes was desired.

Steady-state conditions were studied in another group of goats. The goats were injected every 24 hr over a 5-day period and sacrificed 4 hr after the last injection. The intestinal tract was sectioned as before and the concentration of the yttrium-90 in the various locations was determined. A true steady state could not be achieved in dogs because of the feeding regimen that had been adhered to in the related dog studies. The following approach was made to this problem and, for purposes of this experiment, was called steady-state conditions for the dog. The radioactivity was mixed with the contents of one can of dog food and fed at the same time daily to each animal. The dogs were killed after the fifth feeding. The same volume from a common stock solution of yttrium-90 was administered each day.

D. *Experimental results*

1. *Single dose to dogs.* The dogs in this group actually received varying amounts of radioactivity at the time of dosing, but for convenience of expression the results have been calculated on the basis of a 1-mc dose of yttrium-90. All but three of these animals were killed serially at intervals during the 24 hr following dosing. The remaining animals were killed at 36, 60 and 84 hr after isotope administration. Three per cent of the dose was recovered in the animal killed at 24 hr. Less than 0.02 per cent of the dose was present in the 36-hr dog, and no radioactivity was detected in the 60- and 84-hr animals.

It is interesting to note, in Fig. 2, that there is a rather rapid decrease in concentration of yttrium-90 in the contents of the stomach, with the concentration falling to a low value after 9 hr. It is observed that the decrease in concentration can be described as a pseudo-exponential decrease ($T_{\frac{1}{2}} = 4.6$ hr) during the first 9 hr after administration. The dogs were not given any food after the dose meal but were allowed free access to water. The observed decrease in concentration probably is due to dilution of the stomach contents with water and digestive juices.

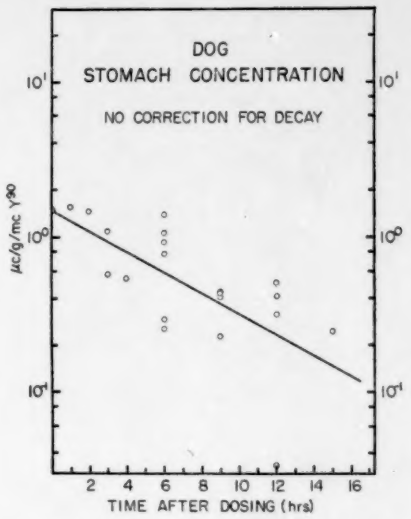

FIG. 2. Yttrium-90 concentration in the stomach of dogs after ingestion of a single dose of the radioactivity, assuming no radioactive decay.

Fig. 3 illustrates the decrease of the total

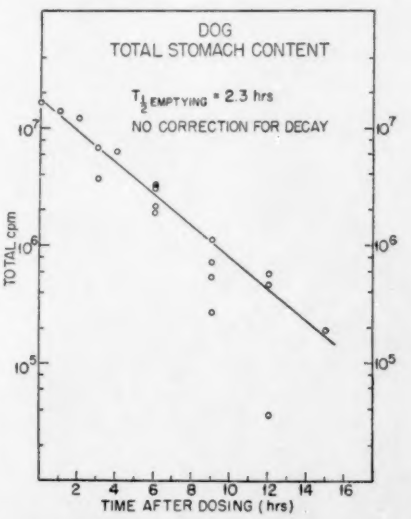

FIG. 3. Radioactivity in the total stomach contents of dogs after ingestion of a single dose of yttrium-90, assuming no radioactive decay.

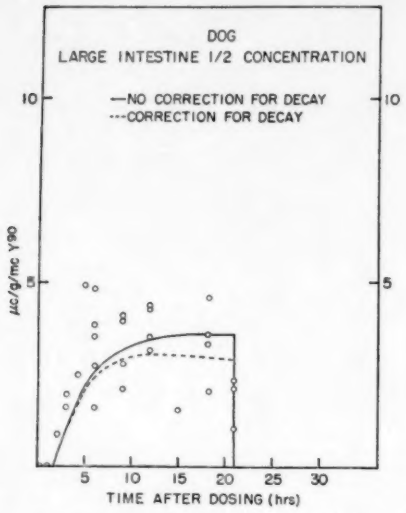

FIG. 4. Concentration of yttrium-90 in the first half of the large intestine of dogs after a single oral dose of the radioactivity. The solid line assumes no radioactive decay.

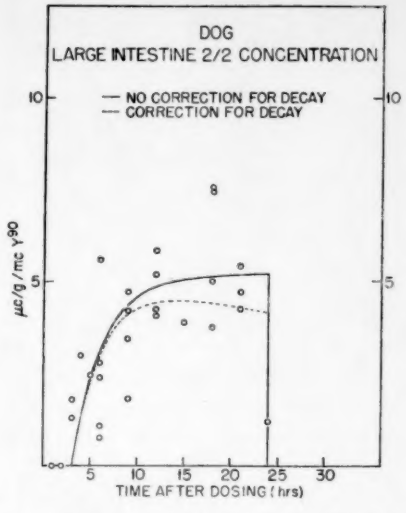

FIG. 5. Concentration of yttrium-90 in the last half of the large intestine of dogs after a single oral dose of the radioactivity. The solid line assumes no radioactive decay.

activity in the contents of the stomach. The slope of this curve is more acute than that noted for the stomach concentration (Fig. 2). This relationship and that of the stomach concentration would be expected because of the decreasing stomach volume with time after ingestion of the radioactive dose meal. An emptying half-time of 2.3 hr is obtained by a "least squares" determination of the pseudoexponential function describing the total stomach activity for a period of 6 hr after dosing. By this time approximately 90 per cent of the activity had passed out of the stomach.

When there is only one bowel movement a day, occurring within several hours after the daily feeding, the build-up of activity in the lower bowel is an approximate reflection of the emptying rate of the stomach. If it is assumed that the time for the dose wave to reach the LI 1/2 and LI 2/2 is 2 and 3 hr, respectively, the build-up in these compartments will be dependent upon the emptying rate of the stomach ($T_{\frac{1}{2}}$ emptying equals 2.3 hr). Selection of an average maximum concentration value for the two compartments from the experimental data (LI 1/2 = 3.6 μc/g per mc Y^{90} ingested; LI 2/2 = 5.2 μc/g per mc Y^{90} ingested) allows calculation of the build-up of radioactivity in these compartments (Figs. 4 and 5). The result is in fair agreement with the visual build-up as indicated by the points obtained in the concentration studies. For purposes of discussion it is also assumed that the LI 1/2 empties completely at the twenty-first hour and the LI 2/2 empties completely at the twenty-fourth hour after feeding. Essentially no activity was observed in any of the dogs killed after 24 hr.

By graphical integration of Figs. 2, 4 and 5, it was found that the stomach, LI 1/2, and LI 2/2 received respectively 8.6 μc hr, 51.8 μc hr, and 78.0 μc hr exposure per millicurie yttrium-90 ingested (Fig. 6). By use of equation (1), a 25-mc dose would result in a 205-rad dose to the stomach, 1235 rads to LI 1/2, and 1864 rads to the second half of the large intestine. This is in good agreement with the radiation dosage determined by direct measurement (Table 2).

2. *Steady state in dogs.* The over-all effects of dilution by digestive fluids and concentration by absorption of nutrient materials and water are clearly shown in Fig. 7.

The two dogs in this group differed in the

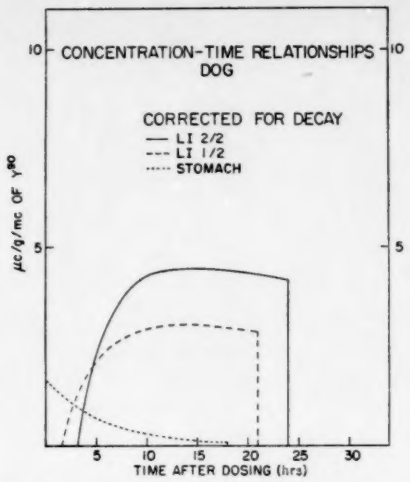

FIG. 6. Concentration–time relationships in the digestive tract of dogs after ingestion of a single dose of yttrium-90.

gross appearance of their bowel movements. Dog *A* was passing firm, well-formed stools, while dog *B*'s bowel movements were much softer in consistency. The concentration of the yttrium-90 in the various parts of the tract is as expected in two such animals.

In passage from the stomach to the cecum in dog *A*, an increasing relative concentration

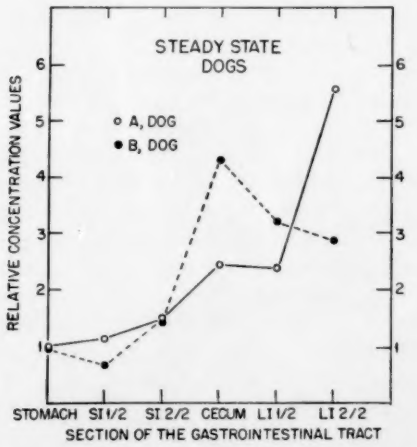

FIG. 7. Relative concentration of unabsorbed radioactivity in the gastrointestinal tract of the dog. These animals were killed after the fifth daily dose.

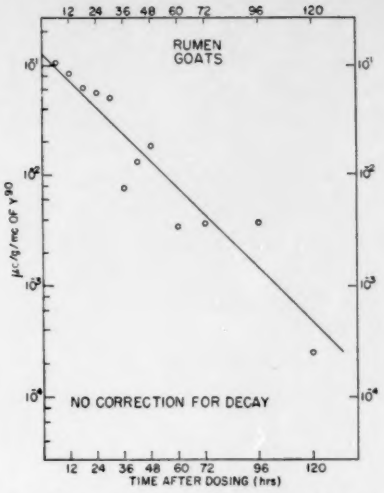

FIG. 8. Concentration of radioactivity in the rumen of goats after injection of yttrium-90 chloride into the contents of the rumen, assuming no radioactive decay.

occurred. Little change was noted in the LI 1/2, but the contents of the LI 2/2 exhibited a sixfold increase over the concentration of the stomach.

In dog *B* the contents of the cecum were concentrated more than four times that of the stomach. However, the concentration values of the large intestine were somewhat lower, and the LI 2/2 to stomach ratio was 3.

3. *Single dose to goats.* Each of the twelve goats actually received 2.1 mc of yttrium-90 solution at the time of dosing, but for purposes of expression the results have been calculated on a basis of a 1-mc dose. These animals were then serially sacrificed at intervals up to 120 hr after dosing.

Fig. 8 presents a scattering of values about the "least squares" determined line. No attempt has been made to standardize the normal variations in rumen content between the various-sized animals (weight range from 60 to 137 lb; average 83 lb). The slope of the line indicates an emptying half-time of about 15 hr.

Figs. 9 and 10 are semilog plots of the concentration values of the injected goats for the last third of the large intestine and terminal colon respectively. Disregarding the 6- and 12-hr values, at which times the concentration

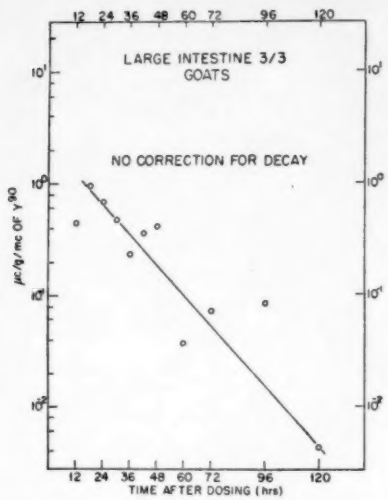

FIG. 9. Yttrium-90 concentration, assuming no radioactive decay, in the last third of the large intestine of goats after injection of the dose solution into the contents of the rumen.

wave of radioactivity had not reached these organs, one obtains an emptying half-time of 15 hr. The slopes of all these lines are comparable and give good evidence that the emptying rate of the rumen is the controlling factor for the entire digestive tract of the goat.

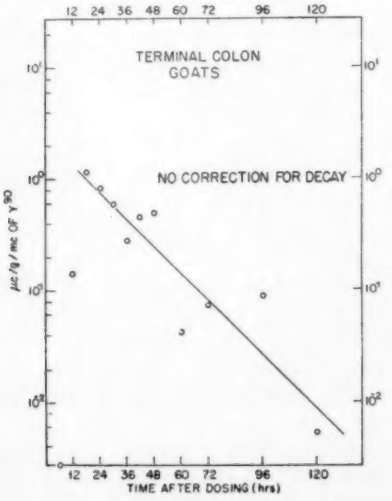

FIG. 10. Concentration of yttrium-90, assuming no radioactive decay, in the terminal colon of goats after injection of the dose solution into the contents of the rumen.

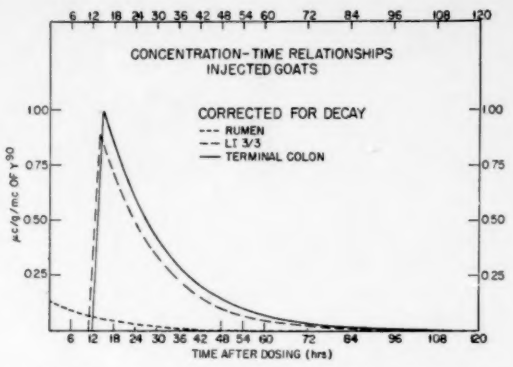

FIG. 11. Concentration–time relationships in the digestive tract of goats after injection of a single dose of yttrium-90 into the contents of the rumen.

Fig. 11 shows the concentration-time relationships of the rumen, LI 3/3, and the terminal colon. The position of the lines was determined from the "least squares" treatments in Figs. 8, 9 and 10 with appropriate correction for decay. It has been assumed that there is a 3-hr build-up time in the lower bowel as the wave of activity arrives. It is felt that this assumption will give a reasonable approximation for the increasing concentration due to mixing that occurs with the contents already present in the lower portions. Dehydration increases as the mass of ingesta progresses through the lower bowel.

There is an immediate exponential decrease in rumen concentration starting at the time of injection with an emptying half-time of about 15 hr. It is estimated that the peak concentration values are reached in the LI 3/3 at about 14 hr and in the terminal colon at about 15 hr. After this maximum concentration has been attained, these lower bowel spaces empty with a half-time the same as the rumen, i.e. approximately 15 hr. Graphical integration of these various curves results in an estimated dose per 25 mc of yttrium-90 to the rumen of 50 rads; LI 3/3 of 340 rads; and terminal colon of 428 rads, which correlates closely with direct measurement, especially the colon values.

4. *Steady state in goats.* As previously described, the goats received five daily injections

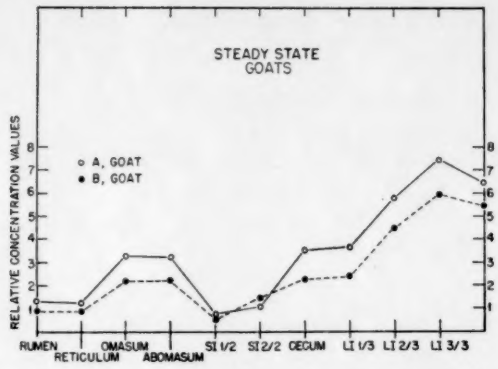

FIG. 12. Relative concentration of unabsorbed radioactivity in the gastrointestinal tract of the goat. These animals were killed 4 hr after the fifth daily injection of the dose into the contents of the rumen.

and were killed 4 hr after the last injections. Fig. 12 is a presentation of the relative concentrations at that time.

In passage from the rumen to the omasum a $2\frac{1}{2}$- to 3-fold increase in concentration occurred. As the ingesta passed through the first half of the small intestine, there was enough dilution (probably from digestive juices) to lower the concentration value below the original rumen concentration. In the second half of the small intestine, it was noted that some increase in concentration took place, probably caused by absorption of nutrients. From the second half of the small intestine there was a continuing increase in concentration with an apparent decrease in the terminal colon. It should be pointed out that this decrease is probably not a true value for the terminal colon, since the final dose (fifth) wave had not yet reached this section in the 4-hr interval. Approximate corrections for this terminal colon value are developed in the discussion section.

IV. DISCUSSION

A. *Comparison of direct dosimetric measurements and concentration–time calculations*

The radiation dose to the intestinal mucosa has been estimated by two different methods. A comparison of results is presented in Table 6. In general it is noted that the agreement is excellent and lends credence to the validity of both procedures.

The agreement for the goat was good for the terminal colon, but the dosimetric value for the rumen was high. This may have been caused by heterogeneity in the rumen contents since the dosimeter was placed in a position (ventral rumen surface) that would tend to give exposure from the more highly concentrated liquid.

B. *Dog data*

The change in concentration of radioactivity in the intestinal contents during passage through the tract is demonstrated in the steady-state studies (Fig. 7). Even with the passage of soft stools, there was a threefold increase in concentration (dog *B*). This compares with the sixfold increase noted for the dog with firm, well-formed stools (dog *A*). Since either type of passage is within the range commonly seen in the dog, perhaps an average over-all increase in concentration of 4.5 is reasonable. It is realized that the variety and digestibility of other diets may cause a variation in this ratio. Certainly, under conditions of constipation or severe diarrhea, this magnitude of concentration change would be altered.

Based on observations of time of residence of ingesta in the stomach and large intestine, it seems that hold-up time in the latter may be of considerable importance in governing the dose received by the large intestine as compared with the stomach. The data from present studies, however, are not adequate to permit quantitative evaluation of this factor.

G. *Goat data*

It is of great interest that the emptying half-time of the various compartments under study in the goat was observed to be about 15 hr. Previous work by other investigators on passage time in the goat[4] and other ruminants[5] has indicated a longer emptying time. These investigators have used stained food particles or plastic beads in their determinations. In general, particles of smaller size pass through the digestive tract faster than larger ones.[6] Since all the data in this report were obtained with soluble yttrium-90 chloride, the observed faster emptying rate is not too surprising, as the more fluid ingesta would be expected to traverse the tract faster than the coarser food particles.

The data clearly indicate that, in the goat, the rumen is the rate-determining organ for elimination. After the initial concentration wave from the rumen reached the lower compartments, the emptying rate for these compartments was the same as that observed for the rumen, about 15 hr.

The changes in concentration of radioactivity during passage through the tract resulting from physiological processes are shown in the steady-state studies. It is interesting to compare the concentrations in the terminal colon and in the rumen. Consideration of the data in Fig. 11 and of the experimental procedures leads to the conclusion that the value shown for the terminal colon is low because the wave of radioactivity from the final injection had not reached the contents of the terminal colon by the time of sampling, 4 hr after the last injection.

A correction should be made for the 4-hr decrease in rumen concentration and for the peak concentration time in the terminal colon. The rumen concentration would be about 83 per cent of the value immediately after injection. It is observed (Fig. 11) that the peak terminal colon concentration occurs about 15 hr after dosing. The goats were killed 4 hr after the last dose, or 28 hr after the fourth dose. Since the increased concentration wave from the last dose had not yet reached the terminal colon, the radioactivity present was a reflection of the fourth dose wave. When this is taken into account, the maximum terminal colon concentration is estimated to be approximately twice that noted at sacrifice time.

If one applies these correction factors, the ratio of concentrations in the terminal colon to the rumen becomes 8.9 instead of 5.4. It should be remembered that this reflects the over-all effect of concentration by absorption of nutrient materials and dehydration.

Since it has been shown that the rate of passage through the terminal colon is approximately the same as through the rumen, then any difference in radiation dose received should be accounted for by differences in concentration. For example, if the values of dosage to the terminal colon and rumen in Table 6 are corrected for decay, the ratio of the calculated values of dosage to the terminal colon and rumen becomes 9.7, which compares well with the value of 8.9 as determined independently.

D. *Species comparison*

Table 6 presents data for species comparison. It is assumed, for purposes of this discussion, that the rate of passage and dynamics of digestion are similar for man and dog. In these species, the lower large intestine (LI 2/2) is the rate-controlling compartment for elimination.

The goat data clearly show a different situation; here the rate-controlling compartment is the rumen. It was observed that the rates of passage of ingestion in all other parts of the digestive tract were either approximately the same or less than that of the rumen.

It is of interest to note that there was a greater increase in concentration in the goat (8.2 to 9.7 times) than in the dog (3 to 6 times) as the ingesta passed through the respective digestive tracts. Since the emptying half-times of the rumen and the terminal colon of the goat are comparable, the increased dose to the lower tract must be largely a reflection of increased concentration of radioactivity in the intestinal content due to dehydration and nutrient absorption.

The condition in the dog is somewhat different. Here the LI 2/2 received more than eight times the stomach dose while the increase in concentration was observed to be only from threefold to sixfold. Thus, undoubtedly, the increased length of time the ingesta remained in the large intestine contributed considerably to the increased dose.

The maximum permissible concentration (MPC) for yttrium-90 can be calculated readily from the data for the different physiological states. It is noted that 4 μc and 17 μc of yttrium-90 for the dog and goat, respectively, will deliver a 300-mrad dose to the critical organ, the lower large intestine. This is true for animals that have normal bowel movements (e.g. in the dog this is defined as daily passage of formed stools). The dogs used in diarrhea studies (nonformed stools) revealed that 12 μc of yttrium-90 would be required to equal the MPC. On the other hand, 1 μc of yttrium-90 would have delivered, on the average, 300 mrads

Table 6. *Species comparison*

Item	Dog		Goat		Man
Compartment of longest hold-up time	Large intestine		Rumen		Large intestine
Concentration increase from stomach (rumen) to terminal colon	3 to 6		8.2 to 9.7		3 to 6*
	Dosimeter	Conc.-time	Dosimeter	Conc.-time	
Dose in rads/mc of Y^{90}	Stomach 8.4	8.2	Rumen 3.8	2	—
	LI 1/2 47.0	49.0	Term.		—
	LI 2/2 75.0	75.0	colon 17.0	17	—
Calculated μc Y^{90} to equal MPC†					
Physiologic state					
Normal	4		17		~ 4*
Diarrhea	12		—		~12*
Constipation	1		—		~ 1*

* Estimated value for man.
† Calculated on 300 mrads/week basis.

to the dogs in the constipation group. This twelvefold difference between the diarrhea and constipation groups emphasizes the importance of the physical state of the bowel passages upon the dose delivered to the critical organ.

E. *Extrapolation to man*

The dog data readily lend themselves to estimation of the effects that might be expected in man.

The lower large intestine of man is the critical portion of the gastrointestinal tract for most unabsorbed radionuclides considered in NBS Handbook 69. One pound (~0.3 lb dry weight) of canned commercial dog food was fed each day to the dogs with normal bowel movements. The food intake of man varies greatly from individual to individual. However, it was found at the time of sacrifice that the average weight of ingesta in the LI 2/2 of the normal dogs was approximately 100 g. This is not greatly different from the 150-g weight in the lower large intestine of man used in determining the MPC's.[1] Since the dose rate is proportional to the concentration of radioactivity, and assuming that the dynamics of the human and dog digestive tracts are similar, it appears that a direct extrapolation of dog data to man would not be out of order. Thus it is predicted that 4 μc of yttrium-90 would deliver 300 mrads to man's lower large intestine. This compares with the maximum permissible intake of 3.1 μc of yttrium-90 as recommended by NBS Handbook 69.[1]

The variation in delivered dose with different physiological states is of extreme importance. Because of the wide differences in dose that would occur as a result of diarrhea and constipation, the establishment of a specific value as the maximum permissible concentration appears to be of limited value unless qualifications are made for the commonly encountered deviations from normal. It should be pointed out that the diarrhea and constipation groups of dogs used in this work were not the extremes observed for these physiological states. A greater spread in doses would be expected in more marked diversions. Also, one must keep in mind that the dogs in the normal group routinely defecated the dose meal at about 24 hr

after ingestion. Any individual who would not excrete the dose meal until a later time would receive a correspondingly larger dose to the critical organ. The data clearly show the value of maintaining a state of diarrhea after ingestion of a nonabsorbed β-emitting radioisotope such as yttrium-90.

It is of interest to compare the relative dose to bone and gastrointestinal tract from chronic ingestion of fallout strontium-90. It has been estimated that a strontium-90 body burden of 100 $\mu\mu$c/g of calcium maintained over a 70-year lifetime could deliver 8–21 rads to the skeleton. If it is assumed that an average daily intake is 500 mg of calcium and that 400 $\mu\mu$c of strontium-90/g of calcium will, on account of discrimination, maintain the previously mentioned body burden, then it can be calculated that the yttrium-90 daughter will deliver approximately 0.4 rad to the gastrointestinal mucosa over the 70-year period. Thus, for chronic ingestion, it is seen that the exposure to the gut from the yttrium-90 daughter is small as compared with the exposure to bone from the strontium parent. It should be noted, however, that this finding does not apply to short-term ingestion where other nonabsorbable radioactive materials may contribute heavily to the intestinal dose.

Acknowledgments—The writers are indebted to Dr. C. HAROLD STEFFEE for interpretations of pathologic material.

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THE METABOLISM OF INDIUM AFTER ADMINISTRATION OF A SINGLE DOSE TO THE RAT BY INTRATRACHEAL, SUBCUTANEOUS, INTRAMUSCULAR AND ORAL INJECTION*

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Abstract—Distribution and excretion of indium after four routes of administration were studied by counting the γ-activity of tissues from rats injected with tracer amounts of indium-114*m*. In 4 days over half the administered dose was absorbed from the injection sites in all but the oral route where absorption was in the order of 0.5 per cent of the administered dose. A general distribution of the isotope was seen after all routes studied, with kidney, spleen, liver and salivary glands containing higher concentrations. Pelt, skeleton and muscle contained the largest absolute amounts of indium, and of these three skeleton had the highest concentration.

Excretion of indium, with the exception of the oral route, was essentially independent of route of administration, occurring with an early rapid phase and a long slow phase which approximated a single exponential after about 20 days. At 30 days after administration 33–40 per cent of the administered dose had been eliminated from the body in urine and feces. Urinary excretion data after all but the oral route were so similar that they were treated statistically as one population and may form the basis for a reliable bio-assay method utilizing analysis of daily urine samples.

I. INTRODUCTION

With the fairly recent admission of indium into the group of industrially important metals has come the problem of both chemical and radiological health hazards. Milling procedures with the resulting dusts, grindings and filings as well as chemical procedures involving solutions or sprays of various indium salts provide many possible routes for contamination of workers and for absorption of the element. To help evaluate these hazards a limited number of laboratory studies on the distribution and excretion,[1–6] and the toxicity,[7–11] of indium have been made in recent years. The presentation to follow will include the results of some of these studies, and will primarily involve quantitative distribution and excretion data from the single intratracheal, oral, subcutaneous and intramuscular injection of indium-114*m* to rats. A possible bio-assay method for determining body burden following accidental exposure is also indicated.

II. MATERIALS AND METHODS

A. *Isotope preparation*

The indium used in this work was reactor produced $_{49}In^{114m}$ in stable carrier described previously.[1,2] From the stock shipments four different systems were prepared immediately prior to injection: (1) an indium hydroxide suspension‡ formed by neutralizing the acid trichloride with NaOH; (2) an indium citrate complex formed by adding an excess of

* This paper is based on work performed under contract with the United States Atomic Energy Commission at The University of Rochester Atomic Energy Project, Rochester, New York.

† Public Health Service, Las Vegas, Nevada.

‡ For the sake of brevity the hydrous indium oxide solution is assumed to be, and is referred to as, indium hydroxide, but its true chemical form has not been studied.

crystalline citric acid to the $InCl_3$ solution and neutralizing with NaOH; (3) the acid trichloride solution itself, diluted with distilled water; and (4) the trichloride solution with additional indium trichloride carrier diluted with distilled water.

B. *Isotope administration*

Four routes of administration were studied; (1) intratracheal intubation (hereafter designated IT) of the hydroxide and of the citrate complex; (2) subcutaneous injection (SQ) of the citrate complex to the hind leg; (3) intramuscular injection (IM) of the trichloride to the muscle of the hind leg; and (4) gavage (oral) using the trichloride solution with additional carrier (stable indium) at two different concentration levels. When necessary, animals were anesthetized with Evipal®† or with 50% CO_2/O_2 prior to injection.

The gravimetric levels of indium injected were 1.2 mg indium/kg body weight in the SQ route, 0.33 mg indium/kg in the IT route, and 0.38 and 0.80 g indium/kg via the oral route. The highest level of radioactive isotope injected in any study was 56 μc/kg of body weight.

C. *Animal maintenance*

Wistar-derived (Rochester strain) male rats weighing 150 g were used for the IT route while females of the same strain and weight were used for all other routes. Animals from which excreta were collected were housed in metabolism cages of the TUTTLE and BAXTER type[1,12] allowing urine and feces to be collected separately with a minimum of cross contamination. All animals were fed Purina Fox Chow in meal form *ad libidum*.

D. *Sacrifice and autopsy*

Sacrifice and dissection was performed according to the procedures described in detail in a previous report.[1]

E. *Counting*

γ-Counting of all samples was accomplished with an overall efficiency of 2.13 per cent as measured by comparison with a cesium-137 and an iodine-131 source. Description of the equipment and general methods employed will be found elsewhere.[1,2]

Counting rates of the samples were tested for significance at the 5 per cent level using the accepted

Table 1. *Mean indium content of injection site and of body burden (body tissues excluding injection site) expressed as percentage of administered dose**

Intratracheal intubation (IT)						
Sacrifice day	2	4	8	16	35	64
Injection site	56.54	40.61	22.19	12.81	13.14	5.62
Body burden	26.18	36.29	46.35	44.76	36.78	34.11
No. of animals	4	4	2	3	4	5
Subcutaneous injection (SQ)						
Sacrifice day	2	4	8	16	37	
Injection site	12.46	14.44	11.80	10.98	4.27	
Body burden	63.33	61.62	55.92	44.89	43.68	
No. of animals	2	2	2	2	2	
Intramuscular injection (IM)						
Sacrifice day	2	4	8	20	32	60
Injection site	—	38.67	30.16	25.90	23.30	9.75
Body burden	—	45.74	49.70	39.76	36.87	33.68
No. of animals	—	3	3	3	3	3
Oral administration						
Sacrifice day	1	4–5	6–8	17–19		
GI tract	97.29	0.49	2.52	0.02		
Body burden	2.34	0.38	0.39	0.10		
No. of animals	1	3	5	3		

* Individual values for the IT and SQ routes are reported elsewhere.[1,2]

† Hexobarbital sodium. Winthrop Laboratories, N.Y.

Table 2. Distribution of indium-114m in rat tissues expressed as mean percentage of body burden per organ and per gram wet weight
(Ranges are given for the intramuscular route to indicate the type of variations involved between animals)

Tissue	Day	Intratracheal		Subcutaneous		Intramuscular		
		per organ	*per g*	*per organ*	*per g*	*per organ*	*per g*	*per g ranges*
Liver	2	11.54	1.60	16.80	2.38	—	—	—
	4	24.25	3.69	17.04	2.32	19.84	3.57	2.96–3.87
	8	12.66	1.86	13.38	1.54	19.98	2.98	2.55–3.59
	18	16.51	1.90	10.16	1.18	12.93	1.84	1.65–2.17
	34	9.57	0.95	6.60	0.82	10.01	1.43	1.23–1.64
	62	8.41	0.70	—	—	7.90	1.15	0.97–1.49
Kidney	2	6.19	4.16	6.95	5.75	—	—	—
	4	7.14	4.96	6.98	5.03	6.50	5.20	4.84–5.40
	8	8.05	5.72	7.01	4.86	6.05	4.51	3.96–5.12
	18	7.06	5.18	5.88	4.05	5.33	4.64	4.44–4.86
	34	5.74	3.21	3.12	1.81	5.43	3.87	3.73–4.10
	62	3.17	1.44	—	—	4.12	3.57	2.29–2.74
Spleen	2	1.18	2.44	1.83	2.92	—	—	—
	4	1.71	4.35	1.74	3.05	1.00	2.88	2.36–3.50
	8	2.14	2.68	2.52	4.49	1.26	3.67	3.04–4.49
	18	2.97	3.55	2.12	3.68	1.40	1.99	1.23–2.72
	34	3.04	4.89	1.15	1.88	2.19	2.04	1.54–2.75
	62	1.99	2.70	—	—	1.44	2.84	2.12–3.28
Salivary glands	2		2.56		1.29		—	—
	4		2.48		1.46		1.97	1.79–2.08
	8		2.65		1.68		2.16	1.79–2.50
	18		2.14		2.18		2.05	1.29–2.87
	34		2.61		2.20		2.90	2.77–3.13
	62		2.55		—		2.56	2.36–2.68
Pelt	2	13.37	0.50	20.26	0.79	—	—	—
	4	15.65	0.63	21.16	0.81	22.58	1.09	0.91–1.38
	8	14.46	0.65	21.69	0.79	18.97	0.85	0.77–0.89
	18	17.25	0.58	18.85	0.76	25.76	0.98	0.94–1.00
	34	21.53	0.76	30.60	0.69	24.70	0.90	0.84–0.97
	62	24.24	0.44	—	—	21.03	0.70	0.36–0.89
Muscle	2	12.30	0.29	16.66	0.41	—	—	—
	4	15.07	0.39	16.31	0.39	12.08	0.41	0.37–0.43
	8	14.37	0.45	19.64	0.38	17.18	0.50	0.45–0.55
	18	18.01	0.34	23.57	0.45	18.68	0.52	0.48–0.60
	34	22.35	0.38	25.30	0.37	19.51	0.47	0.35–0.55
	62	25.92	0.27	—	—	22.03	0.45	0.41–0.52
Skeleton	2	18.83	0.99	18.19	1.34	—	—	—
	4	22.32	1.05	26.70	1.46	20.98	1.72	0.77–3.13
	8	18.42	0.99	25.61	1.79	19.61	1.16	0.48–1.87
	18	22.56	1.17	27.04	1.85	23.31	1.25	1.08–1.35
	34	22.95	0.90	26.38	1.15	24.32	1.33	1.17–1.62
	62	20.58	0.67	—	—	26.77	1.00	0.87–1.28

Table 3. *Distribution of indium-114m with indium-115 trichloride added after oral administration to the rat*
(Body burden is expressed as per cent of administered dose. Organs and tissues are expressed as percentage of body burden per organ)

Dose of stable indium (g/kg)	Day	Body burden	Percentage of body burden per organ					
			Kidney	Spleen	Liver	Skin	Bone	Muscle
0.80	1	2.34	10.34	0.39	3.50	5.26	60.00	4.14
0.38	4	0.65	11.85	trace	14.77	19.08	26.46	10.15
0.38	5	0.062	0.0	13.14	20.83	0.0	17.63	15.70
0.38	5	0.42	11.14	3.08	14.93	20.83	20.62	13.27
0.80	6	0.32	11.23	1.81	21.92	9.40	21.82	5.88
0.80	7	0.36	11.24	2.56	25.98	17.63	23.18	5.45
0.38	7	0.51	2.14	trace	3.70	13.45	11.11	7.80
0.38	7	0.16	4.21	trace	7.55	20.76	29.56	24.53
0.80	8	0.61	12.56	2.57	24.10	7.88	28.01	9.30
0.38	17	0.11	0.0	0.0	trace	23.85	33.03	13.76
0.38	18	0.12	trace	0.0	9.92	23.97	28.93	21.49
0.38	19	0.060	trace	trace	trace	trace	trace	trace

method for calculating standard deviation of a counting rate. (Two standard deviations above background ranged from 20 to 30 counts/min.) When counting rates were not significant by this test the indium content was reported as "trace" unless it was to be used in finding a total such as total excretion. Then the actual amount of indium represented by the low counting rate was included in the total.

F. *Collection and notation of data*

The time schedules for sacrifice and the number of animals involved at any point are shown in Table 1. Since the sacrifice schedule and general experimental plan of the first three routes were similar, it is feasible to make a comparison of the distribution and excretion data from them. This is done in Table 2. Data from the oral route, because of its different sacrifice schedule and its markedly different absorption and excretion pattern, are presented alone in Table 3. Because the metabolism of both the "hydroxide" and the citrate complex were so strikingly similar[1] the two have been combined here as one IT group. On this basis comparison of metabolism of the soluble trichloride with that of the citrate and "hydroxide" is justifiable.

Tissue levels of indium are expressed as percentage of administered dose corrected for radioactive decay rather than of recovered dose since the fraction of administered indium actually recovered was high and relatively constant. Average percentage recovery for the four routes was as follows: 94.2 per cent for IT, 96.3 per cent for SQ, 92.8 per cent for IM and 90.0 per cent for the oral route.

Many of the distribution data are expressed as percentage of body burden per gram of tissue. *Body burden* is considered here to be the amount of indium present in the entire body with the exception of the injection site organ or area, whereas the term *body content* (used in Fig. 1) refers to the indium content of the whole body including injection site. Thus, the body burden represents the amount of indium presented to the circulation which is remaining in the body tissues. Percentage of administered dose in any tissue can be determined by simple arithmetic from the data given.

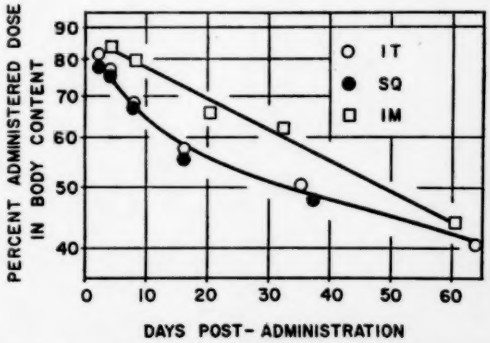

FIG. 1. Per cent of administered dose in the body (body content) of rats receiving indium-114*m* by intratracheal, intramuscular and subcutaneous injection.

Table 4. Excretion of indium-114m after four routes of administration to the rat expressed as percentage of administered dose

Day	Total cumulative excretion (individual averages)				Daily feces values (weighed averages)				Feces/urine ratios (weighted averages)		
	IT	SQ	IM	oral	IT	SQ	IM	oral*	IT	SQ	IM
1	6.80	7.62	2.42	trace				trace			
2	9.88	12.35	4.61	36.26	3.26	3.60	1.53	36.11	6.37	1.83	2.01
3	13.46	16.38	6.96	73.52	3.54	3.67	1.69	37.16	8.23	5.73	2.96
4	17.22	20.19	9.31	84.98	3.40	3.28	1.76	11.10	8.72	6.43	3.52
5	21.22	23.25	11.47	86.32	3.60	2.58	1.59	1.34	9.00	5.61	3.18
6	24.92	25.46	13.25		3.60	2.00	1.47		9.00	4.65	3.42
7	29.48	25.45	15.01		2.64	1.54	1.24		6.77	4.16	3.35
8	31.12	28.97	16.29		2.07	1.27	1.10		5.31	4.38	3.55
9	32.61	30.79	17.47		1.66	1.06	0.92		4.37	3.92	3.07
10	34.42	33.12	18.66		1.47	0.94	0.82		3.97	3.76	2.83
11	35.90	34.20	19.66		1.28	0.86	0.79		3.56	4.10	2.72
12	37.18	35.26	20.71		1.10	0.77	0.75		3.24	3.50	3.00
13	38.38	36.12	21.66		0.98	0.69	0.75		2.97	3.28	3.12
14	39.41	36.98	22.60		0.87	0.66	0.67		2.90	3.00	2.79
15	40.35	37.83	23.41		0.78	0.63	0.59		2.78	3.32	2.46
16–20	44.18	40.05	27.13		0.67	0.53	0.51		3.12	2.84	2.44
21–25	46.95	42.59	30.11		0.46	0.37	0.43		2.41	2.40	2.51
26–30	49.15	44.69	32.61		0.34	0.30	0.37		2.21	2.59	2.53
31–40	52.86	—	37.22		0.27	—	0.32		2.13	—	2.40
41–50	55.65	—	40.76		0.20	—	0.24		2.47	—	2.19
51–60	58.07	—	44.16		0.17	—	0.22		2.28	—	2.18

* These are individual averages, not weighted averages.

III. RESULTS

A. *Absorption of administered dose*

Indium was readily absorbed from the injection site after IT, SQ and IM administration. Table 1 shows that at 4 days over half the administered dose had been absorbed or excreted. After the eighth day injection site values decrease at a rate which is somewhat the same for all three routes of administration. The initial rapid loss from the lung can be partly attributed to ciliary action of the tracheal epithelium since early gastrointestinal (GI) tract[1] and fecal excretion values (Table 4) were high. These differences in initial rate of injection site clearance are reflected in the plot of body burden data presented in Fig. 2. Although data for the first few days are widely scattered they result in a body burden on the eighth day of 56 per cent for the SQ route, 50 per cent for the IM and 46 per cent for the IT route. The influence of these differences in initial absorption is reflected in the distribution data to be presented.

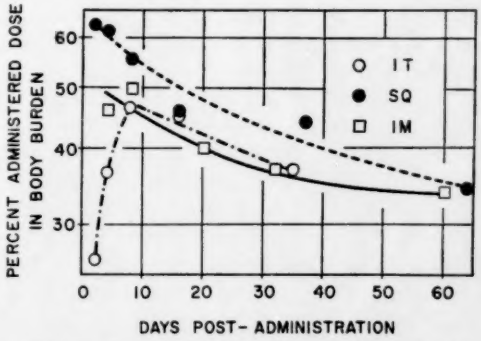

FIG. 2. Per cent of administered dose in the body burden (body content excluding injection site) of rats receiving indium-114*m* by intratracheal, intramuscular and subcutaneous injection.

Absorption after the oral route was very low. No more than about 2 per cent of an ingested dose was absorbed through the intestinal tract and one-fourth of this amount may be a more realistic figure. Data from all animals having oral administration of indium are included in Table 1 since no difference in absorption or distribution was observed with differences in stable indium dose (Table 3).

The negligible absorption of indium from the GI tract validates the assumption that after intratracheal intubation the indium brought up through the trachea and swallowed contributes very little to the body burden of these animals.

B. *Distribution of indium*

Table 2 shows that indium was distributed generally throughout the body and with few exceptions the distribution pattern was the same after all routes studied. Although some organs concentrated the isotope more than others, it would be difficult to choose a "critical" organ on the basis of indium concentration. The organs and tissues can be arranged in a continuous series in order of their decreasing indium concentration, with kidney, spleen, liver and salivary glands heading the list in that order during the first 2 weeks. After about a month spleen and salivary glands tended to have a higher concentration than kidney and liver. The carcass (pelt, skeleton, muscle) contained by far the greatest absolute amount of indium and on this basis can be considered the major site of deposition. Of these three tissues, skeleton contained the highest concentration, even higher than that in liver in some cases after 30 days. Although there were no startling differences in distribution pattern between the four routes studied, some minor variations may be noted. In most cases the concentration of indium in organs and tissues of rats injected subcutaneously dropped off faster than in the same organs or tissues from animals in the other groups. This may be an effect of the more rapid initial absorption seen after SQ administration which was discussed previously.

Distribution data from the oral route (Table 3) show a somewhat similar pattern to those described above. However, the indium absorbed from the GI tract was found at such low absolute levels in the tissues that very accurate γ-counting was not feasible. There appears to be no detectable "carrier effect" on absorption or distribution. In general, kidney, spleen and liver contained the highest concentrations, and pelt, skeleton and muscle were again the indium reservoirs. Kidney may have accounted for a slightly higher proportion of the body burden during the first week than it did after the other injection routes.

C. *Excretion of indium*

The excretion pattern of indium was seemingly independent of route of administration, with some minor exceptions (Fig. 3). Each curve represents the average daily urine or feces indium content from animals maintained in metabolism cages for the duration of each study and, therefore, the number of values averaged for each curve varies from eight for the IM route to two for the SQ route. The daily values used to obtain these curves are weighted averages, or, the average of the mean value of the group on that day, on the day preceding and on the one following it. Each of the three routes illustrated shows an initial rapid phase of excretion in the first 10–20 days followed by a progressively slower phase. The half-times of these two distinct components can be calculated, although they are somewhat arbitrary because the period of data collection was short compared to the

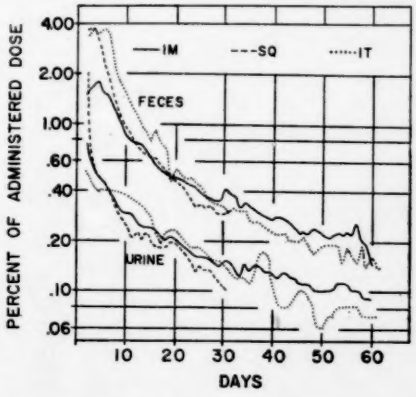

FIG. 3. Daily excretion of indium-114*m* as percentage of administered dose in urine and in feces of rats after intramuscular, subcutaneous and intratracheal injection.

half-time of the processes involved in the slower phases. The rapid phases of total excretion occurred with half-times of about 3 days for the SQ, 5 days for the IT and 6.5 days for the IM group. The long slow phase was calculated to occur with a half-time of between 94 (SQ) and 126 (IM) days. It is this latter phase that should indicate the magnitude of half-time for excretion to be expected in a long-term chronic exposure.

Initial excretion of the isotope plays an important part in determining the differences in total cumulative excretion. Data presented in Table 4 show that after about 20 days the daily output of indium in IT, IM and SQ groups was essentially the same, yet due to the differences in initial daily output half the administered dose was excreted in 32 days by the IT group, 44 days by the SQ group and 80 days by the IM group.

The urinary excretion pattern after oral administration is complicated by the low absorption from GI tract mentioned earlier. Six per cent and 3 per cent of the absorbed dose appeared in the urine on the first and fourth days after oral administration while 1 per cent was excreted at these times after IT administration. This may be a metabolic difference but is most likely a result of the inaccuracies of the low-level counting found in the orally injected animals. Eighty-six per cent of the administered dose, or almost 100 per cent of the recovered dose was eliminated in 4 days via the feces. An important implication of these excretion data is that elimination of ingested material through the GI tract occurs very rapidly and unless the original dose is quite high analyses of the urine will show only trace amounts of indium. No significant amounts of the isotope were found in either urine or feces after 4–5 days although the dose contained 3.5×10^5 counts/min.

IV. BIO-ASSAY METHOD FOR INDIUM

The similarities in daily urinary excretion of indium after IT, SQ and IM administration are such that the individual urine values from each route can be combined on each day and treated as one. The essentially "normal" distribution of the combined daily urine values allows them to be treated statistically as one population. When the daily means of the individual values

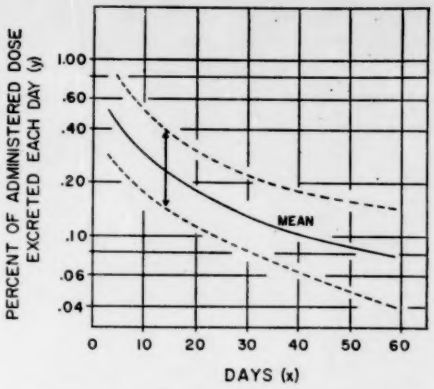

FIG. 4. Approximate 75 per cent tolerance interval around the mean daily urinary excretion of indium-114*m* (percentage of administered dose) after three routes of administration.

are plotted as a weighted average, a smooth curve may be fitted to them (solid line, Fig. 4). The 75 per cent tolerance interval[13] was calculated at the 95 per cent confidence level to determine the limits which include at least 75 per cent of a normally distributed population with confidence 95 per cent. The smooth curves through these limits are represented by the two broken lines in Fig. 4. From this graph the single dose received by an individual may be estimated through the simple relationship that on x day of collection the indium content of the urine excreted represents y per cent of the original dose. An approximation of the prevailing body burden or body content can then be made from this estimate of the initial dose by referring to Figs. 1 and 2.

To determine the magnitude of an exposure to man by this method necessitates the common assumption that data obtained from study of the rat can be applied to man. It also implies that intratracheal intubation is a valid index for evaluation of an inhalation hazard. A comparison of the distribution and excretion studies using polonium-210 in the rat after both intubation[14] and inhalation[15] shows that distribution of this material, once it has entered the circulation, may be quite similar for the two routes. The greatest differences arise in behavior of the material in the lung. A recent study of the

metabolism of indium sesquioxide after inhalation in the rat[3] showed an over-all distribution pattern that was not greatly different from that seen after intubation of the "hydroxide" and the citrate complex discussed here. Since daily urine samples were not taken throughout the inhalation study a comparison of rate or level of urinary indium excretion cannot be made. Nevertheless, this bio-assay method may be of some considerable use in evaluating an inhalation exposure to man until data with more direct application can be obtained.

Note

No MPC values were calculated from this work since it was felt that a multiple dose study was needed wherein continuous exposure conditions would be more closely approximated. Such a study has now been completed. With distribution and excretion data from both single and multiple exposures to the rat MPC values for air and for water were calculated and were compared to those recommended in the NBS Handbook 69.[16] A report of this multiple dose study along with these calculations will be found in the University of Rochester Atomic Energy Project Report UR-554.[6]

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THE CESIUM–POTASSIUM DISCRIMINATION RATIO

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Abstract—An investigation has been made of the γ-ray activity of foodstuffs eaten by Toronto residents in the spring of 1959, and a comparison made with the activity of the residents themselves. From this, it is calculated that the Cs–K discrimination ratio, $(Cs/K)_{food}:(Cs/K)_{body}$, is 1:3.

INTRODUCTION

DURING the last few years, concern has been expressed over the increase of radioactive fission products in the human body, and in human food. Most scientific work on this subject has been carried out with Sr^{90}, a bone seeker and possible cause of somatic damage. Cs^{137} has received less attention, though it may cause genetic damage.

It is of considerable value to know whether, and to what extent, the body discriminates against these fallout products. Clearly if the body has a mechanism which rejects cesium as compared to the chemically similar element potassium, the danger to the body due to contamination of the food supply will be reduced. Once again, the discrimination against cesium has received less attention than the discrimination of the body against strontium, although it is nowadays relatively easy to measure the cesium-137 content of the body and of foodstuffs.

Basing their measurements on the activities of milk and people, ANDERSON[1] has concluded that the proportion of cesium to potassium in the body is about twice that in food, while BOOKER[2] finds that the specific cesium activities in milk and people are approximately equal.

The present paper gives the results of experiments on the Cs–K ratio of food-stuffs consumed by Ontario residents in the spring of 1959 and the Cs–K ratio of the residents themselves. From these figures a discrimination ratio for the Cs–K pair of elements is established.

NOMENCLATURE

It should be noted at this point that there appears to be some confusion over the use of the term "discrimination ratio". LIBBY[3] means by the discrimination factor against Sr the ratio of the specific amount of Sr^{90} ($\mu\mu c$/g Ca) in foodstuffs to that in, for example, the human body. This figure is greater than unity in all biological transfers studied. However, ANDERSON[1] uses the term discrimination factor (for cesium) to mean the ratio $(Cs/K)_{man}$ to $(Cs/K)_{milk}$, i.e. the inverse use to that of LIBBY. MILLER and MARINELLI[4] define an assimilation ratio to denote the ratio of $(Cs/K)_{body}$ to $(Cs/K)_{urine}$, while BOOKER[2] uses an observed ratio for the quantity $(Cs/K)_{body}/(Cs/K)_{milk}$.

In fact the amount of any isotope X in an organ depends essentially on four factors[5]:

(a) the daily intake in food, X_{food} (disregarding inhalation);
(b) the fraction which passes from the gastro-intestinal tract to the blood (f_{1X});
(c) the fraction passing from the blood to the organ (f_{WX});
(d) the time the isotope remains in the body (associated with the effective half-life τ_{eX}).

In the cases of cesium-137 and potassium, τ_e is equal to the biological half-life τ_X for all practical purposes, and f_W is equal to unity as the whole body is the critical organ. It follows that the amount of potassium in the body at equilibrium is given by

$$K_{body} = K_{food} \cdot f_{1K} \cdot \tau_K / 0.693$$

and the ratio of cesium-137 to potassium in the body is similarly

$$(\mathrm{Cs/K})_{\mathrm{body}} = (\mathrm{Cs/K})_{\mathrm{food}} \cdot (f_{1\mathrm{Cs}}/f_{1\mathrm{K}}) \cdot (\tau_{\mathrm{Cs}}/\tau_{\mathrm{K}})$$

in the simple case of equilibrium. The ratios of the f_1 term and the τ term are here called the assimilation ratio and the retention ratio, respectively. It will be seen later that this equation is complicated in practice by the fact that equilibrium conditions do not exist. In the following discussion, we will follow the usage of LIBBY[3] and speak of a Cs–K discrimination ratio defined as $(\mathrm{Cs/K})_{\mathrm{food}}/(\mathrm{Cs/K})_{\mathrm{body}}$. That is,

$$\text{d.r. (Cs/K)} = 1/\text{assimilation ratio (Cs/K)} \times \text{retention factor (Cs/K)}.$$

If this discrimination ratio is less than unity, then the Cs/K ratio in the body is higher than that in food.

ACTIVITY OF FOOD-STUFFS

At the present time the radioactive contamination of food-stuffs is at a relatively low level, and as a result the activity is difficult to distinguish above background. We have carried out measurements using a 5-in. NaI crystal inside a "steel room"[6,7] with 8-in. thick walls. Normally a 10-lb lot of food-stuffs has been used, and counting carried out over a period of an hour or more.

The question arises as to which food-stuffs to test, as clearly a comprehensive programme is impossible. Human food may be divided into the following rough categories:

(1) Above the ground vegetables: (i) of which the whole or main part is eaten, e.g. cauliflowers; (ii) of which only a part is eaten, e.g. fruits, cereals.
(2) Below the ground vegetables, e.g. potatoes, carrots.
(3) Animal products.

It is believed that cesium is taken into plants via the roots only to a small extent[8] whereas it is relatively easily ingested through the leaves.[9] Thus one would expect that leafy foods might have a different cesium-137 content from root crops. Again, once vegetable matter is taken into animals, a relative concentration of cesium, or the reverse, may take place. Thus we tested all four categories of food. Within the different categories, food-stuffs which were relatively rich in potassium were chosen, as it would be expected that the concentration of cesium in these would also tend to be high, owing to the similar distributions of cesium and potassium in living material.

The foods tested were bought from shops in Toronto. As the purchases were made in the spring of 1959 many of the foods were imported and thus the results are not necessarily typical of foods grown in Ontario but rather typical of food eaten in Ontario—the significant quantity in this test.

The cesium–potassium ratios of the foods were measured by comparing the relative sizes of the 0.66 MeV and 1.46 MeV photopeaks and applying corrections for isotopic content, nuclear decay scheme and differential response of the detector, as outlined in a previous paper.[7] No attempt was made, except in the case of the milk, to obtain absolute figures for the cesium and potassium content.

The results for the Cs/K ratios, expressed in cesium units (C.U., $\mu\mu$c of Cs^{137} per g of K) are shown in Table 1, together with the intake of potassium per day from various sources for a person on a normal Canadian diet.[10] It is clear from the table that dairy products are the main source of cesium-137 in Canada.

ESTIMATION OF DISCRIMINATION RATIO

Using the methods outlined in a previous paper[7], the γ-activity of twenty Ontario residents with no history of radioactive contamination was measured in the spring of 1959. These persons had an average weight of 165 lb and an average age of 36 years. The result for the average Cs/K ratio, 57 C.U., is in good agreement with that obtained on thirty medical students in the fall of 1958.[7]

Combining the results of these two sets of measurements, the average potassium content of the body is 0.21 per cent of the body weight, with a standard deviation of 0.023 per cent, and the Cs^{137} burden is 8.5 mμc with a standard deviation of 2.6 mμc.

A straightforward comparison of the Cs/K ratios in food and in humans will give a first approximation to the discrimination ratio, but a correction should be applied because the Cs/K ratio in food has been changing with time. Although no figures are available for most foods, milk has been studied for some years, and its rising activity has here been taken as typical of that of other food-stuffs. To a sufficient degree of approximation, the rise in activity may be taken as linear with time, with zero

Table 1. *Intake of cesium-137 and potassium*

Product group	Cs–K ratio (C.U.)	Daily potassium intake (g)	Daily cesium intake ($\mu\mu$c)
1. (i) Cauliflower	0 ± 6	0.78*	0.78*
(ii) Oranges	5 ± 5	0.47	1.0
Bananas	0 ± 1		
Bread	23 ± 6	0.22	5.1
2. Potatoes	0 ± 2	0.41	0.0
Carrots	2 ± 2	0.78*	0.78*
3. Beef	67 ± 6	0.40	21.8†
Fish	5 ± 2		
Milk	35 ± 3	1.0	35
		3.28	63.7

Weighted average cesium intake: 63.7 $\mu\mu$c. Cs–K ratio: 19 C.U.

* From *all* vegetables except potatoes.

† Assumed that one-fifth of potassium intake is via fish.

activity in early 1955.[2] The rise of strontium activity shows a similar pattern.[11]

The effect of this changing cesium activity may be found by setting out and solving the relevant differential equation. At any time the rate of increase of activity in the body

$$(d\mathrm{Cs}/dt) = \text{rate of intake } (f_{1\mathrm{Cs}} I' t) - \text{rate of excretion } (0.693\ \mathrm{Cs}/\tau_{\mathrm{Cs}})$$

where Cs is the amount of Cs^{137} in the body at any time, I' is the rate of increase of cesium in the daily food intake, and thus $I't$ is the intake at any time t, and τ_{Cs} is the biological half-life of cesium in the body. The solution of this equation, when $\mathrm{Cs} = 0$ at $t = 0$, is

$$\mathrm{Cs}_t = f_{1\mathrm{Cs}} \frac{I' t \tau_{\mathrm{Cs}}}{0.693} \times \left\{1 - \frac{\tau_{\mathrm{Cs}}}{0.693 t}\left[1 - \exp\left(-0.693 t/\tau_{\mathrm{Cs}}\right)\right]\right\}.$$

It will be seen that after a sufficient time, Cs is increasing linearly with t, but that the body activity lags the food activity by an amount dependent on τ. It is this lag term which is the cause of the necessary correction noted above.

For conditions in which the rate of intake is a constant, and for which equilibrium normally exists, as in the case of potassium, the corresponding equation is

$$\mathrm{K} = f_{1\mathrm{K}} \frac{I \tau_{\mathrm{K}}}{0.693}$$

where K = amount of potassium in the body, I = rate of intake and τ_{K} is the biological half-life of potassium in the body.

Actual values can now be substituted in these equations. The daily intake of cesium-137 is now (spring, 1959) 64 $\mu\mu$c (see Table 1). On the assumption of linear rise with time, with zero activity in the spring of 1955, $I' = 64/4 \times 365$ $\mu\mu$c/day per day. For the other terms, we accept τ_{Cs} as 120 days,[12,13], τ_{K} as 38 days,[5] and I as 3.28 g of potassium per day (Table 1). Then

$$(\mathrm{Cs}/\mathrm{K})_{\mathrm{body}} = (f_{1\mathrm{Cs}}/f_{1\mathrm{K}}) \times 54 \text{ C.U.}$$

As stated above, the average cesium-137 activity of persons measured in Toronto in the spring of 1959 was 57 C.U. Thus

$$\frac{f_{1\mathrm{Cs}}}{f_{1\mathrm{K}}} = \frac{57}{54} = 1.05$$

and the discrimination ratio,

$$(\mathrm{Cs}/\mathrm{K})_{\mathrm{food}}/(\mathrm{Cs}/\mathrm{K})_{\mathrm{body}} = (f_{1\mathrm{K}}/f_{1\mathrm{Cs}}) \cdot (\tau_{\mathrm{K}}/\tau_{\mathrm{Cs}}) = 0.33$$

A check on one part of this calculation may be made by considering the amount of potassium in the body as calculated and as measured. Using values for f_{1K} and τ_K of 0.9 and 38 days,[5] and a daily intake of 3.28 g of potassium, then

$$(K)_{body} = 0.9 \times 3.28 \times 38/0.693 = 162 \text{ g K}$$

The average of the measured amounts of potassium in the bodies of the Toronto residents was 151 g. The difference between these figures gives an idea of the errors involved in this kind of calculation.

It will be seen that the correction term to be applied for nonequilibrium conditions is small. Thus any error in the assumptions about the beginnings of appreciable cesium-137 activity or its linear rise will cause only a very small change in the calculated value of the discrimination ratio.

DISCUSSION

The results, based on the trace quantities of radioactivity absorbed in food, show that the probability of cesium being absorbed into the blood stream is approximately the same as that for potassium—we obtain $f_{1Cs} = 0.95$ as against 0.9 for f_{1K}.[5] However, the greater time that cesium remains in the body has the effect of making the specific cesium activity of the body greater than that of food-stuffs by a factor of 3.

For Ontario residents, we estimate that more than half the cesium-137 is received in milk, and another third in meat products. Other potassium-rich foods have relatively little cesium-137 associated with them. These latter conclusions are very similar to those reached by ANDERSON *et al.*[14] for persons in the U.S.A. at a time when the cesium contamination was very much less than it is now.

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THE SENSITIVITY OF PHOTOGRAPHIC FILM TO 3-MeV NEUTRONS AND TO THERMAL NEUTRONS*

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Abstract—Measurements were made to determine the sensitivity to 3-MeV neutrons and to thermal neutrons of three commercial photographic films (DuPont film types 502, 510 and 606). The films were exposed in the conventional "NBS holders", consisting of Bakelite covered with thin layers of lead and tin. The results of the measurements were compared with results obtained by rough theoretical estimates. The sensitivity of the films to neutrons was found to be relatively low. On the least sensitive film, a photographic density of 0.1 was obtained with a fast-neutron exposure of 6.8×10^{10} n/cm^2. On the most sensitive film, the exposure for the same density was 4.9×10^9 n/cm^2. The ratios of the films' fast-neutron-to-γ-ray sensitivities were found to be roughly equal to the reciprocal of the ratios of their average grain diameters. The thermal neutron sensitivity of the films was of the same order as the 3-MeV sensitivity, and was roughly the same for all three film types. In the few cases in which values for the neutron sensitivity could be obtained theoretically, they showed order-of-magnitude agreement with the experimental values.

1. INTRODUCTION

It is often important to know the error in the photographic dosimetry of X- and γ-rays introduced through the presence of neutrons. A good estimate of this error rests upon an accurate determination of the sensitivity of photographic film to neutrons alone.† But neutron fluxes are always accompanied by γ-radiation. Since, in terms of flux, film is considerably more sensitive to γ-radiation than to neutrons, its response to γ-rays easily masks its neutron response. Therefore, accurate information on the neutron sensitivity of photographic film can be obtained only if the effects of neutrons and of γ-rays can be separated. This requires either very "clean" neutron fluxes, or accurate means for measuring γ-ray exposure in the presence of neutrons.

So far, only relatively little has been published on the neutron sensitivity of commercial photographic film. Early experimental studies were aimed toward the use of such film for personnel dosimetry of neutrons.[1–4] These and later studies led to the use of commercial film, with cadmium or rhodium foils as intensifiers, for monitoring thermal neutrons.[5] The action of fast neutrons on commercial photographic film has not been sufficiently enhanced, as yet, to make fast-neutron dosimetry by densitometric means feasible. Nevertheless, for certain procedures in X- and γ- ray dosimetry with commercial film, the photographic effect of fast as well as of thermal neutrons is not negligible.

An attempt to determine the extent of the difficulty introduced by fast neutrons in photographic X- and γ-ray dosimetry was published by Tochilin *et al.*[6] in their paper on the photographic effect of charged particles and neutrons. They exposed three types of commercial photographic film to rather broad fast neutron spectra with considerable γ-ray background, and, for lack of a better instrument, determined the γ-ray exposure from the response of the film type

* This work was supported in part by the U.S. Atomic Energy Commission. Preliminary experimental results were reported in 1958 at the meeting of the International Radiation Research Society (W. L. McLaughlin and M. Ehrlich, *Rad. Res.* **9**, No. 1, 1958).

† Throughout this paper, the term sensitivity (or photographic sensitivity) is defined as the reciprocal of the exposure required for a given optical density.

whose neutron sensitivity they expected to be particularly small. They obtained experimental data on the sensitivity of the films to fast neutrons relative to their γ-ray sensitivity, using the raw fast-neutron data without a correction for the γ-ray contamination; they found their results to be comparable with theoretical estimates made by utilizing the experimental data for the proton sensitivity of the same film types. Their later studies were mainly directed toward a comparison of the shapes of density vs. exposure curves for exposures to radiations of different specific ionization, rather than toward the determination of absolute or relative sensitivities.[7]

During the past five years, we have tried to obtain more accurate information on the absolute fast- and thermal-neutron sensitivities of commercial photographic film, and on the ratio of its neutron-to-γ-ray sensitivities. In Section 2 of this paper, the radiation sources and exposure techniques used in some of the experiments are discussed. In Section 3, the mechanisms of the interactions of neutrons with photographic film are reviewed and an outline is given of how the photographic effect of neutrons was estimated theoretically. The experimental data are discussed in Section 4 in the light of results obtained by the simple theoretical estimates.

2. EXPERIMENTAL TECHNIQUE

2.1. 3-MeV *neutrons**

The 2-MeV Van de Graaff at the National Bureau of Standards (NBS) was used to produce neutrons by the $D(d, n)$ He^3 reaction on a deuterium target occluded in aluminum. In order to keep the γ-ray background low, a deuteron energy of only 0.32 MeV was employed. The energies of the neutrons produced by these deuterons in the thick target were, on the average, 3.08 MeV in the direction of the deuteron beam, and 2.50 MeV perpendicular to this direction.[8] Fig. 1 shows the all-aluminum target assembly. For the sake of reproducibility and stability, the film holder supports, designed for exposures of film in the NBS-type holder[9] at 0°, 45° and 90°, were welded onto the actual target structure (T in the drawing). The mass of the assembly was kept close to its structural minimum. Neutron flux was measured by means of an associated-particle counter and a "long counter", calibrated with a Pu–Be source

* All operations connected with the production and dosimetry of the 3-MeV neutrons were carried out by the staff of the NBS Neutron Physics Section.

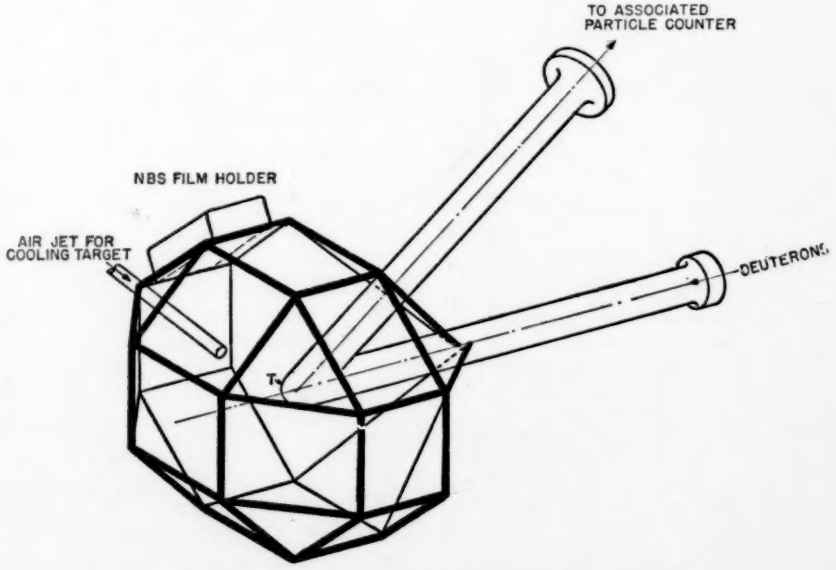

FIG. 1. Exposure geometry for neutron work at 3 MeV.

of known emission rate. The accuracy of the neutron-flux measurement is estimated to have been not greater than 15 per cent.

The films selected for the experiment were of the same type and batch that had been employed for the measurement of total γ-ray exposure at one of the United States' atomic test series. They were the DuPont film types 502, 510 and 606, all contained in the DuPont dental-size packet, type 553. No batch-to-batch comparison studies of the neutron sensitivities have been performed, so far; therefore, the results should not be taken to hold quantitatively for any but the behavior of the particular batches on which the experiments were carried out.

Table 1 lists the approximate chemical composition of the three types of film, and gives the approximate average diameters of their silver halide crystallites (grains) before and after development in Kodak Liquid X-ray Developer. The weight percentages shown for gelatine and for the film base and the thicknesses of the emulsion components were determined at NBS from a number of samples of the three film types. The average grain sizes before development were furnished by the research department of DuPont; the average grain sizes after exposure to fast neutrons and after development were estimated in our Laboratory by a rough microscopic procedure.

Total exposures ranging from about 6×10^8 to 5×10^{10} n/cm^2 were given to the DuPont type-553 packets enclosed in NBS-type film holders, consisting of an electronic equilibrium layer of Bakelite, covered with layers of lead and tin. The target-to-film distance was about 9 cm. Film data were obtained in both the zero and ninety-degree positions of the film holder supports. Since, in spite of the difference in neutron energy at the two positions, the data agreed within the accuracy limits of the experiment, they were combined, and the experiment is briefly referred to as the "3-MeV experiment".

At one position, the film type 606 was exposed without the other two films of the type-553 packet, in order to check on a possible influence on its response by the relatively more neutron-sensitive film type 510. Within the accuracy of the experiment, no difference in response was found.* The degree

* A marked influence of the proximity of film type 510 on the response of type 606 had been noticed previously in radiation fields associated with atomic explosions, possibly due to activation of the silver halides by the thermal neutron components of that radiation field.

Table 1. *Chemical composition and average grain sizes of films used*

(a) *Chemical composition**

Chemical compound and approximate weight per cent of its elements	Thickness (g/cm^2) for DuPont film type		
	502	510	606
Gelatine (C, 50%; H, 7%; N, 17%; O, 25%; S, 1%)	0.00273	0.00491	0.00083
Silver halide (Ag, 57.4%; Br and other halogens, 42.6%)	0.00210	0.00435	0.00077
Cellulose acetate (base) (C, 50%; H, 5.5%; O, 44.5%)	0.0269	0.0259	0.0187

(b) *Average grain sizes*

Condition	Average grain size (μ) for DuPont Film Type		
	502	510	606
Before processing	1	0.1	0.3
After exposure to fast neutrons and processing in X-ray developer	2.5	0.8	0.5

* Excluding trace elements.

of thermalization of the 3-MeV neutrons in the Bakelite layer of the NBS holder was determined by means of counts on gold foils placed on the outside and inside of the holders; the number of thermal neutrons activating the gold foils inside the holder was found to be about 0.5 per cent of the number of incident fast neutrons.* A possible influence of the lead and tin filters of the NBS holders on film response to neutrons was checked by exposing the Bakelite portions of the holders with and without their metallic covers. No difference was observed within the experimental precision limits, estimated to have been about 10 per-cent. The gamma-ray background at the site of the film holders was surveyed with a proportional counter of relatively low neutron sensitivity,[10] and was found to be less than or equal to 1 per-cent of the first-collision neutron dose at the same point.

2.2. *Thermal neutrons*†

NBS film holders, loaded with films of the same types and batch numbers as those used for the fast-neutron experiment, were exposed at Oak Ridge National Laboratory (ORNL) to thermal neutrons from a reactor thermal column. Similar exposures were given at Los Alamos Scientific Laboratory (LASL) to the DuPont film packets, type 554, containing the film types 510 and 606, of a batch different from the one used for the other phases of the experiment. All film types of a particular packet were exposed simultaneously and remained in contact with each other and in their original packet until processing. No attempt was made to eliminate or correct for possible effects of activated film samples on each other.

The thermal fluxes at the exposure positions were known from conventional measurements, such as from determinations of the activity of Au^{198}. The associated gamma-ray exposures were determined at LASL with ionization chambers and at ORNL with photographic film, both enclosed in lithium shields. The thermal-neutron exposures ranged from about 1×10^8 to 4×10^{12} n_{th}/cm^2. The gamma-ray exposure associated with an exposure of 1×10^8 n_{th}/cm^2 was roughly 2.9 mr at LASL and 65 mr at ORNL. Because of the higher gamma-ray background at ORNL, the ORNL data will be used only for the film type 502 which was not included in the type-554 packet exposed at LASL. For the other two film types, the LASL data will be used.

2.3. *Photons*

For a comparison of the sensitivity of the films to neutrons and to high-energy photons, Co^{60} γ-ray exposures administered at NBS were used in conjunction with the NBS and ORNL neutron exposures. The high-energy X-ray exposures used with the LASL thermal-neutron exposures were administered at LASL.

2.4. *Processing*

All films exposed to 3-MeV or thermal neutrons were processed at NBS together with the associated photon-exposed film samples. Kodak Liquid X-ray Developer was used for 5 min at 20°C. In order to illustrate the influence of processing technique on the results presented in this paper, another experiment is briefly discussed in Section 4.3, in which the difference between density vs. exposure curves obtained after processing in Kodak D-76 Developer and in Kodak Liquid X-ray Developer are shown both for exposures to fission-neutrons and to photons.

3. THE INTERACTION OF PHOTOGRAPHIC MATERIALS WITH NEUTRONS

3.1. *Mechanisms of interaction*[11–14]

The more important interactions of fast neutrons with the components of a photographic film are elastic scattering on hydrogen and carbon, and inelastic scattering, leading to γ-ray emission as well as to activation of silver and bromine. Preliminary estimates showed that the predominant portion of the photographic effect was due to proton recoils from elastic scattering of fast neutrons on the hydrogen of the emulsion, of the film base, and of the surrounding materials, such as the paper of the film packet and the plastic of the film holder.

Fast neutron–proton elastic scattering is essentially isotropic in the center-of-mass system, and thus—in the laboratory system—shows a cosine distribution in the proton recoil angle up to 90°, and is zero for larger angles. (This means that the cross-section has a maximum in

* This is not necessarily a negligible amount. No comparison between film response with and without holder is available for 3-MeV neutrons. However, for 0.5-MeV neutrons, for which the thermalization effect should be more pronounced, experiments showed that the difference between the film responses with and without holders was within the 10-per-cent precision limit of the experiment.

† All operations connected with the production and dosimetry of thermal neutrons were carried out by the staff of the Biology Division of LASL, and of the Health Physics Division of ORNL.

the forward direction, and goes to zero at 90°; no protons recoil backwards.) Gelatine, film base, paper and plastic have somewhat similar hydrogen densities. For the present experiments, the range of the most energetic of the recoiling protons was smaller than the combined thickness of these substances. Therefore, the exposure of the photographic films took place under conditions not far from proton equilibrium.

In the case of thermal neutrons, conditions are much more complicated. Thermal neutrons are captured both in bromine and in silver, and the radioactive isotopes produced in this process have rather complicated decay schemes, involving a number of high- and low-energy γ-rays, as well as β-radiation, and having half-lives from a few seconds to several hundred days. Matters are further complicated by the fact that the high-energy β- and γ-rays produced during the decay of the activation products originate solely in the silver halide grains, which are not large enough to establish conditions of electronic equilibrium in silver halide.

Besides the thermal-neutron capture reactions, there are also (n_{th}, p) reactions in carbon and particularly in the nitrogen of the gelatine. Preliminary estimates showed the film exposure due to the protons from nuclear reactions induced by thermal neutrons to be small compared to that due to the decay products from the thermal-neutron activations; the effect of these protons will not be considered in the present estimate.

3.2. *Semiquantitative estimate of sensitivity*

In order to calculate the neutron sensitivity of commercial photographic film, one would have to compute the number of charged particles and photons produced by the neutrons in the film emulsion and its surroundings, determine the fraction of the number and energy of the particles absorbed in the individual silver halide grains, and estimate the resulting film density from the number of grains that become developable. The calculation of the energy transfer to the silver halide grains is particularly difficult, because of the variation in the size of the grains in a given commercial emulsion. The variation in grain size also complicates the optical conditions that have to be considered in a determination of photographic density from the number of developed emulsion grains. Furthermore, since the inherent sensitivity of the individual grains (a quantity proportional to the energy required for developability) is not constant even for any given grain size, the difficulties involved in the task of computing the film sensitivity in detail become insurmountable. However, in some instances one can obtain rough estimates for film sensitivity to neutrons by using available experimental information on the sensitivity of the film to the interaction products, and on the number of developed grains corresponding to a given photographic density. In the following, procedures for working out such estimates will be outlined in some detail.

Fast neutrons. Only neutron–proton elastic scattering will be considered. The experiments can easily be carried out under conditions of proton equilibrium, and the number of recoiling protons is usually small compared to the total number of hydrogen atoms present. For an order-of-magnitude calculation, one may further assume that all protons recoil at the same angle, namely the angle for which the differential elastic scattering cross-section has its average value (angle $\phi_{\text{av}} = 45°$).

Let N_H be the number of atoms per cm² in the entire emulsion (two layers for double-layer emulsions), let $\sigma_{\text{sc}}(E_n)$ be the elastic (n,p) scattering cross-section (in cm²/atom) for neutrons of energy E_n MeV; let, furthermore, n_{gr} be the number of grains per g/cm² of the emulsion (here called the "linear" grain density). Then, under conditions of proton equilibrium, the fast-neutron flux required for a particular film density is given by

$$F_{n,\text{fast}}(p) = N/[f_{\text{av}} N_H \sigma_{\text{sc}}(E_{n,\text{fast}}) n_{\text{gr}} R(\phi_{\text{av}})] \quad (1)$$

where N is the number of developed grains per cm² required for that density, $R(\phi_{\text{av}})$ is the range of protons in g/cm² in the 45° direction, and f_{av} is the average fraction of the number of grains encountered that is actually made developable. In intermediate steps not shown here a linear relationship was assumed, (a) between the number of grains made developable per incident neutron and the number of grains required for a given optical density, and (b) between the

number of grains required for a given optical density and the neutron flux required for that density. Strictly, this step is justified only if the relations between grain density and optical density and between optical density and neutron flux are linear.

Thermal neutrons. Tables 2, 3 and 4 show the cross-sections for some of the more important modes of interaction of thermal neutrons with photographic emulsions, as well as the energies of the decay products. The type of calculation required for the estimate of the film response to these decay products will now be outlined for the decay of the silver isotopes.

Let N_{Ag} be the number of silver atoms per cm^2 in the entire emulsion; let $\sigma_c(\beta_i)$ be the cross-section for the production of the ith β-ray due to thermal-neutron capture. Furthermore, let us assume that we know from an experiment the quantity $\rho^{0.1}$ in ergs/cm^2, which is the energy density of the monoenergetic electron beam required to produce a photographic density of 0.1 in the particular film. This energy density is assumed to have the same effect as the energy density of a β-ray beam of about the same average β-ray energy ($\overline{E_{\beta_i}}$). Then the thermal-neutron flux $F_{n,\mathrm{th}}(\beta)$ required for a photographic density of 0.1 due to β-rays alone is given by

$$F_{n,\mathrm{th}}(\beta) = (1/N_{Ag})\rho^{0.1} / \sum_i \left[\sigma_c(\beta_i)\overline{E_{\beta_i}}\right] \quad (2)$$

Table 2. Radiations resulting from thermal neutron capture in silver

(Ag^{109}, 48.65%; Ag^{107}, 51.35%)

Type of radiation	Cross-section (barns)	Energy* (MeV)
270-*day* Ag^{110} (2.8 barns) *in equilibrium with* 24.5-sec Ag^{110} (110 barns)		
β-rays (−)	1.6	0.087 (0.023)
β-rays (−)	0.98	0.53 (0.16)
β-rays (−)	1.1	2.12 (0.89)
β-rays (−)	3.3	2.86 (1.8)
γ-rays	3.8	0.656
γ-rays	0.17	0.676
γ-rays	0.34	0.706
γ-rays	0.51	0.759
γ-rays	0.17	0.814
γ-rays	0.98	0.935
γ-rays	0.34	1.389
γ-rays	0.34	1.516
2.3-min Ag^{108} (44 barns)		
β-rays (−)	44	1.77 (0.69)

* The numbers in the parentheses are average values, obtained under the assumption that the spectra involved were allowed.

Table 3. Radiations resulting from thermal neutron capture in bromine

(Br^{79}, 50.57%; Br^{81}, 49.43%)

Type of radiation	Cross-section (barns)	Energy* (MeV)
4.6-hr Br^{80} (2.9 barns) *in equilibrium with* 18-min Br^{80} (8.5 barns)		
β-rays (−)	1.19	1.38 (0.52)
β-rays (−)	6.62	2.00 (0.80)
β-rays (+)	2.29	0.866 (0.38)
Auger electrons	0.45	0.0125
γ-rays	1.19	0.62
γ-rays	2.9	0.048
γ-rays	2.9	0.036
35.9-hr Br^{82} (3.5 barns)		
β-rays (−)	3.5	0.45 (0.14)
γ-rays	3.5	0.54
γ-rays	3.5	0.61
γ-rays	3.5	0.69
γ-rays	3.5	0.72
γ-rays	3.5	0.82
γ-rays	3.5	1.03
γ-rays	3.5	1.30
γ-rays	3.5	1.45

* The numbers in the parentheses are average values, calculated under the assumption that the spectra involved were allowed.

Table 4. Reaction of thermal neutrons with the nitrogen of gelatine

(N^{14}, 99.64%; $N^{14}(n_{\mathrm{th}}, p)C^{14}$)

Type of radiation	Cross-section (barns)	Energy (MeV)
protons	1.75	0.557[15]
β-rays (−) from 5720-year C^{14}	1.75	0.155

This holds under the assumption that the β-ray energy produced per neutron incident on a given square centimeter of area is absorbed within the cylindrical volume defined by this surface area or, at least, that as much β-ray energy is lost out of the cylinder defined by 1 cm^2 of surface area as is gained from the surroundings.

The effect of the capture-γ-rays, as well as the effect of the protons produced in the carbon and nitrogen of the gelatine, was found to be negligible compared with that of the β-rays, and will not be considered in the present estimates.

The calculations of the effects of thermal capture in bromine are similar to those here outlined for silver, and are not discussed separately. The thermal-neutron flux required to produce a given density due to the β-rays from both the silver and the bromine is obtained by having $\sum_i [\sigma_c(\beta_i)\overline{E_{\beta_i}}]$ in the denominator of equation (2) extend over the cross-sections and average β-ray energies of the β-rays both from silver and from bromine.

4. RESULT

4.1. *Density vs. exposure relations*

Figs. 2, 3 and 4 show the net photographic densities obtained on the three film types as a function of neutron flux for exposures to 3-MeV neutrons at NBS. Fig. 5 shows similar data for the film type 502 exposed to thermal neutrons at ORNL. The corresponding data on thermal-neutron exposures of film types 510 and 606 are shown in Figs. 6 and 7; they are based on results obtained from LASL exposures. The ORNL thermal-neutron data for film types 510 and 606 are not shown. In spite of a difference in emulsion batch, they are similar to the LASL data; however, because of the higher γ-ray background, they contain even larger uncertainties and would require larger corrections.

The figures also show photon response curves obtained on film samples processed simultaneously with those used for the respective neutron response curves. The solid lines shown in the graphs were obtained by the method of least-square fit from the experimental points (black dots for neutron exposures, open circles for X-

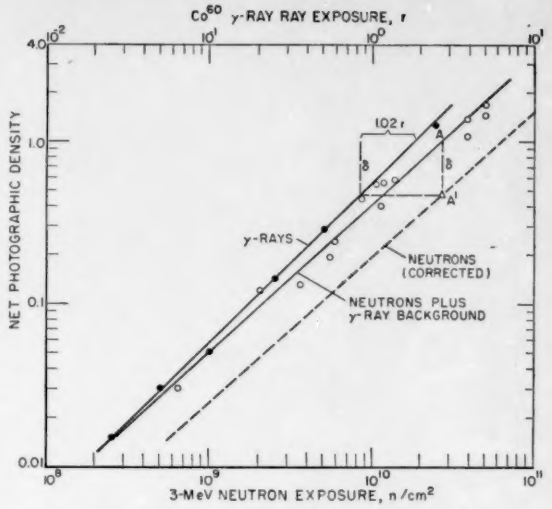

FIG. 2. DuPont film type 502. Photographic density as a function of exposure to 3-MeV neutrons and to γ-rays. Point A on the uncorrected neutron curve goes into point A' on the neutron curve corrected for γ-ray background.

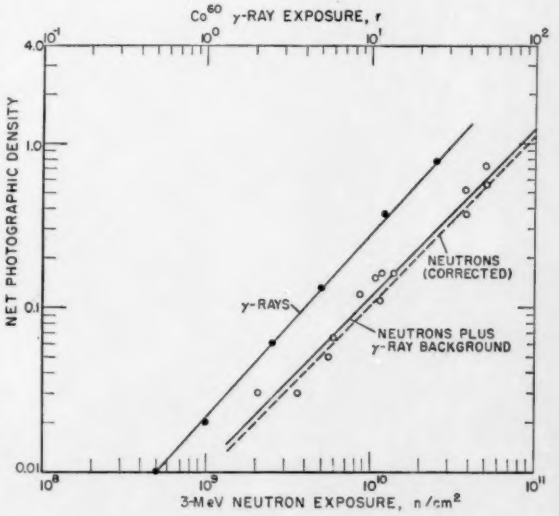

FIG. 3. DuPont film type 510. Photographic density as a function of exposure to 3-MeV neutrons and to γ-rays.

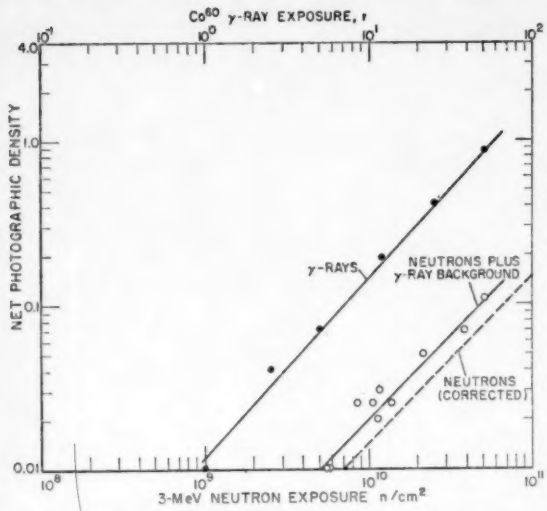

FIG. 4. DuPont film type 606. Photographic density as a function of exposure to 3-MeV neutrons and to γ-rays.

and γ-ray exposures). The dashed curves marked "neutrons (corrected)" were obtained from the curves marked "neutrons plus γ-ray background" by deducting the actually measured γ-ray contamination. The correcting procedure will now be outlined.

Let us follow the steps leading to the correction of the unit density point A in Fig. 2 on the curve marked "neutrons plus γ-ray background". First, assuming that the γ-contamination was exactly 1 per cent of the first-collision neutron dose in tissue-equivalent materials, we determined the amount of γ-contamination in r corresponding to the 3-MeV neutron flux that produced unit density.* Conversion factors of 3.7×10^{-9} rads per n/cm^2[16] and of 0.93 rad/r in tissue were used. For unit density, a γ-ray contamination of 1.02 r was calculated in this way. This exposure was then deducted from the γ-ray exposure required for unit density, by decreasing the abscissa of the γ-ray curve by this value. The corresponding decrease in the ordinate (δ) is the density due to the 1.02 r of γ-contamination. Finally, the value δ was subtracted from the unit ordinate of A, leading to the corresponding point A' on the corrected

* Actually, all that was known from the experiment was that the γ-contamination was less than 1 per cent. Therefore we are over-correcting the curve, and, as a consequence, overestimating the film's neutron sensitivity.

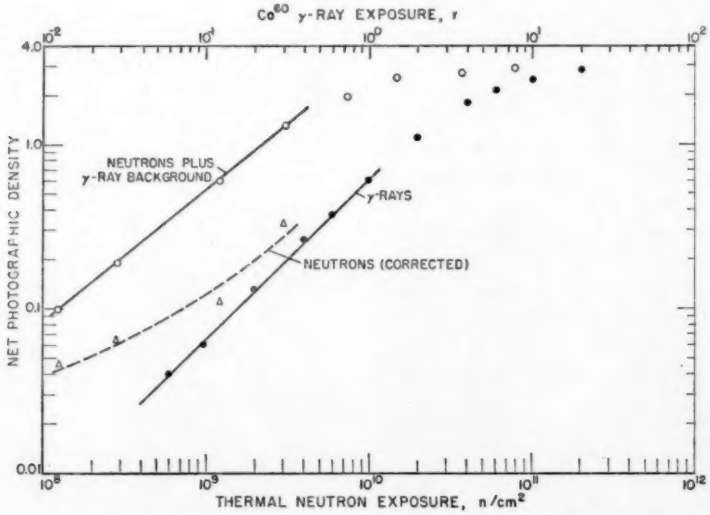

FIG. 5. DuPont film type 502. Photographic density as a function of exposure to thermal neutrons and to γ-rays. The triangles are the result of the correcting procedure.

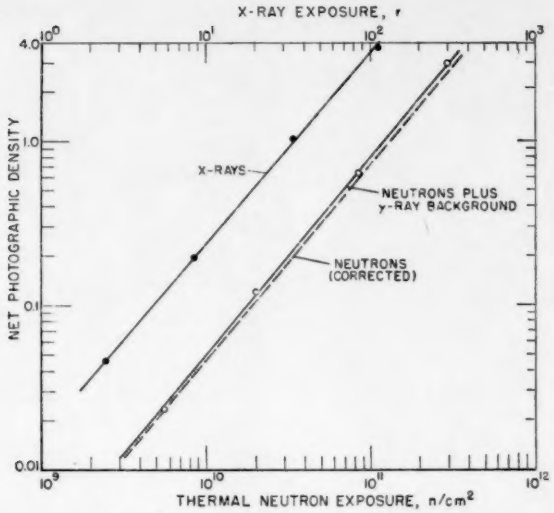

FIG. 6. DuPont film type 510. Photographic density as a function of exposure to thermal neutrons and to X-rays.

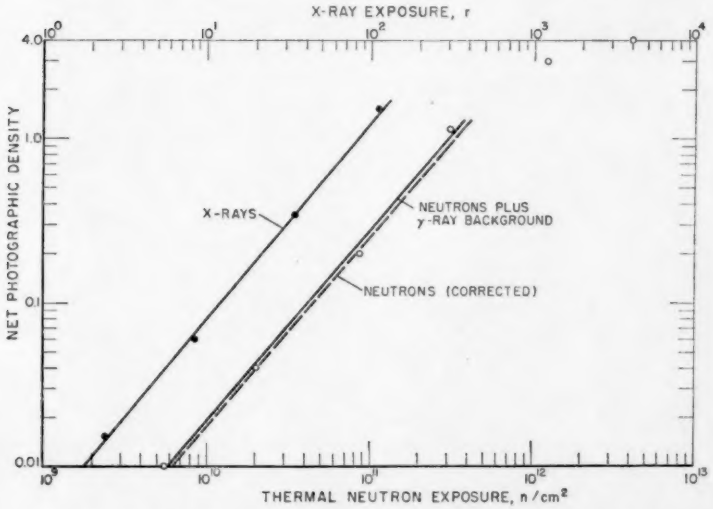

FIG. 7. DuPont film type 606. Photographic density as a function of exposure to thermal neutrons and to X-rays.

neutron curve.* A similar procedure was followed for several points on every uncorrected neutron curve, except that in the case of thermal neutrons the γ-contamination was immediately available in r, which made the first step described for fast neutrons unnecessary.

An inspection of the results of this correcting procedure shows how important it is to keep the γ-ray contamination at a minimum. In the case of the ORNL exposures to thermal neutrons, which were done with a considerable γ-ray background, the uncertainties in the knowledge of the γ-ray contamination seem to have been large enough to cause the corrected points of film type 502 to stray considerably. Unfortunately, this film type was not contained in the film packets exposed at LASL and the results on the thermal-neutron sensitivity of this film, therefore, are even poorer than one would expect them to be on the basis of its high γ-ray sensitivity.†

4.2. *Estimated and measured values of film sensitivity*

Fast neutrons. Table 5 gives a list of the numerical values available for the calculation of the fast-neutron flux required for a given photographic density from equation (1). The hydrogen concentration was computed from the data given in Table 1(a). The linear grain density n_{gr}, defined as the number of grains per g/cm^2 in a filament having the cross-section of an average grain, was determined, under the assumption of essentially spherical grains, as

$$n_{gr} = 3/2\ t_{AgBr}/(t_{em}\ d_{av}\ \rho_{AgBr})$$

d_{av} denotes the average grain diameter before development (in cm), ρ_{AgBr} the density of silver bromide (in g/cm^3), and t_{em} and t_{AgBr} are the area densities (thicknesses in g/cm^2) of the emulsion and of the silver bromide contained therein. This expression for the grain density was tested experimentally on several occasions, and was found to hold fairly well for emulsions with nearly spherical grains.[17] Since X-ray emulsions are ammoniacal, and ammoniacal emulsions are known to have nearly spherical grains,[17] linear grain density should be fairly independent of direction in the X-ray emulsion types 502 and 510; no information is available on the grains of the ciné-positive emulsion type 606.

The range of the protons in the direction of 45 degrees was estimated from the data given for ranges in nuclear-track emulsions by DEMERS[18] as

$$R_{comm} = R_{n.t.}\ \rho_{n.t.}\ A_{comm}^{1/2} / \rho_{comm}\ A_{n.t.}^{1/2}$$

where R_{comm} and $R_{n.t.}$ are the ranges in the commercial and nuclear-track emulsions, respectively, ρ_{comm} and $\rho_{n.t.}$ are the corresponding densities, and A_{comm} and $A_{n.t}$ the corresponding average atomic weights of the emulsions. The quantity f_{av}, the average fraction of the number of grains seen by a proton and actually made developable, is not readily determined for commercial film material. WEBB found that, up to a proton energy of about 7 MeV, all grains along a proton track become developable in an average nuclear-track emulsion.[17] In our experiments, the energies of the recoil protons were much smaller than 7 MeV, and the grains of at least two of the three emulsions used were, on the average, larger than the grains of nuclear-track emulsions. Nevertheless, because of the considerable variation in grain size and the more non-uniform grain sensitivity of ordinary commercial emulsions, it is not justified to assume that all grains along a proton track of

* This correcting procedure implies a linear relation between exposure and density, both for γ-rays and for neutrons. Over the density range of interest, the deviations from linearity of the experimental data (which, on the graphs, would manifest themselves in deviations from unit slope of the log-density vs. log-exposure lines) actually are seen to be quite small.

† In general, if the neutron sensitivity of all three film types were roughly proportional to their γ-ray sensitivity, and the γ-ray contamination were the same for all three of them, the amount of correction for γ-ray contamination and therefore also the error introduced in the correcting procedure due to an incorrect estimate of this γ-contamination, would increase with increasing γ-ray sensitivity of a film. The film type 502, which is the one most sensitive to γ-radiation, actually required the largest correction. The film type 510, when exposed to fast neutrons, is an exception. It is more sensitive to γ-radiation than film type 606, but because of its relatively larger neutron sensitivity it requires a somewhat smaller γ-ray correction than type 606. Consequently, the error introduced in the corrected fast-neutron response curve due to the uncertainties in the amount of γ-ray contamination is smaller for film type 510 than for the other two film types.

Table 5. Numerical values used in the estimate of film response to fast neutrons

Quantity	Film type		
	502	510	606
N_H (atoms/cm^2)	1.14×10^{20}	2.05×10^{20}	3.51×10^{19}
n_{gr} (number per g/cm^2)	~1000	~11000	~3700
$R(\phi_{av})$ (g/cm^2)		$\sim 7.5 \times 10^{-3}$	
$\sigma_{sc}(E_n = 3$ MeV) (barns)		~2.5	
f_{av}	—	0.25	—
N, number per cm^2			
for $D = 0.05$	1.6×10^6	1.2×10^7	1.8×10^7
for $D = 0.10$	3.2×10^6	2.4×10^7	—

one of these commercial emulsions become developable.

A microscopic study of tracks in commercial emulsions is usually not feasible because the emulsions are thin and their grain density is relatively low. However, on DuPont film type 510, which has the highest grain density of the emulsions used in the present study, proton tracks containing up to thirty grains were actually found, whose grain density showed a definite increase toward the end of the tracks. A tilt-stage microscope was not available and it was therefore impossible to do grain counts on tracks with a 45-degree dip angle. However, some tracks parallel to the film base were found to have ranges corresponding to proton energies between 1.5 and 2.0 MeV, and f_{av} was estimated from two of them. According to this estimate, on the average, between one-fourth and one-fifth of the encountered grains of the DuPont 510 emulsion are made developable by the recoil protons produced in the emulsion and its surroundings by an incident beam of 3-MeV neutrons.

The number of grains per cm^2 (N) required for a given film density was obtained experimentally by grain counts both on neutron-exposed and on γ-ray exposed film material whose optical density had been determined. Although it is not evident *a priori* that the results should be the same for neutron- and γ-ray exposures, the values obtained were the same, within the limits of the counting accuracy, for the film types 502 and 606. For film type 510, N tended to be somewhat smaller for neutrons than for gamma rays, indicating that the developed grains might be somewhat larger after proton than after electron exposure. Grain counts were done for optical densities between about 0.03 and 0.15. Within the accuracy limits of the measurements, no deviation from linearity of the relation between grain count and optical density could be observed for these low densities. Within the accuracy limits, the relation between optical density and neutron flux also was linear for this density range. Thus, it is justified to use equation (1) which implies a linear relation between neutron flux and the number of developed grains.*

Table 6 shows measured and calculated results for the three film types. The neutron fluxes required for densities 0.05 and 0.10 are shown in section (a) of the table.† For film types 502 and 606, the calculated values are only relative because no data for f_{av} are available. The absolute value calculated for film type 510, using the f_{av} obtained in the experimental estimate, is seen to be smaller by roughly a factor of 4 than the measured value. Considering the

* Strictly, the fact that some grains in the path of a proton do not become developable makes one suspect that the neutron response is not independent of the rate of neutron incidence. However, rate dependence could only occur if more than one proton recoil were to lose energy in the same grain. This is not likely to happen at the flux densities used in the present experiments.

† These two relatively low densities were chosen for all tabulations of the experimental results because the theoretical estimates were carried out for densities in this range.

Table 6

(a) *Fast neutron flux required for a given optical density*

Film type	Fast neutron flux (n/cm²) for			
	$D = 0.05$		$D = 0.10$	
	Calculated	Measured	Calculated	Measured
502	$8.2 \times 10^8/f_{av}$	2.2×10^9	$1.6 \times 10^9/f_{av}$	4.9×10^9
510	$3.1 \times 10^8/f_{av}$ $\approx 1.4 \times 10^9$	5.0×10^9	$6.2 \times 10^8/f_{av}$ $\approx 2.8 \times 10^9$	9.8×10^9
606	$8.1 \times 10^9/f_{av}$	3.5×10^{10}	$1.6 \times 10^{10}/f_{av}$	6.8×10^{10}

(b) *Ratio, fast-neutron-to-γ-ray sensitivity*

Film type	Reciprocal of measured fast-neutron flux (cm²/n) required to produce the same density as 1 r of γ-radiation	
	from $D = 0.05$*	from $D = 0.10$*
502	$1/(2.5 \times 10^{10})$ [1]	$1/(2.8 \times 10^{10})$ [1]
510	$1/(2.3 \times 10^9)$ [11]	$1/(2.4 \times 10^9)$ [12]
606	$1/(8.9 \times 10^9)$ [2.8]	$1/(9.4 \times 10^9)$ [3.0]

(c) *Ratio of reciprocal grain diameters prior to processing*

Film type	502 510 606
Ratio	1 : 10 : 3.0

* The numbers in the brackets are relative fluxes.

many uncertainties in the estimate, better agreement cannot be expected.

Table 6(b) shows for the three film types the reciprocals of the ratios of the measured fast-neutron fluxes and the γ-ray exposures required for densities 0.05 and 0.10. These quantities are in units of r per unit neutron flux. Since the photographic sensitivity is defined as the reciprocal of the exposure required for a given density, the quantities also represent the ratios of the measured neutron-to-γ-ray sensitivities of these films. The values in the brackets are the ratios of the neutron-to-γ-ray sensitivities for the three film types relative to the ratios of the neutron-to-γ-ray sensitivity of film type 502. Film type 510 is seen to be more than ten times as sensitive as type 502, and film type 606 about three times as sensitive. It is interesting to note that the ratio of the neutron-to-γ-ray sensitivities of the three film types is roughly equal to the reciprocal of the ratio of their grain diameters [see Table 6(c)]. This is not implausible, provided the f_{av} values of a given film type for the proton recoils produced by 3-MeV neutrons in a hydrogenous material are proportional to the f_{av} values for electrons produced by roughly 1-MeV photons. The argument could then go like this: Once a grain is made developable, any additional energy imparted to it is wasted, as far as the photographic process is concerned.

Table 7. Numerical values used in estimate of film response to thermal neutrons

Quantity	Film type		
	502	510	606
$N_{Ag} = N_{Br}$ (atoms/cm^2)	6.73×10^{18}	1.39×10^{19}	2.47×10^{18}
$\rho^{0.1}$ (ergs/cm^2)	0.056	not available	
$\sigma_e(\beta_i)$ (barns)	Ag : 51 barns Br : 14 barns		
$\overline{E_{\beta_i}}$	Ag : 0.74 MeV $= 1.2 \times 10^{-6}$ ergs Br : 0.49 MeV $= 0.78 \times 10^{-6}$ ergs		

Electrons in the energy range considered probably never impart to the grains much more energy than is required for developability (and often less). The recoil protons impart to the grains considerably more energy than the electrons, probably often much more than is required for developability. Since the energy loss of a proton to a grain is essentially proportional to the path length of this proton in the grain (and thus to the grain's linear dimension) the energy expended per developable grain should be inversely proportional to the linear grain dimension. The relation between the number of grains made developable and the developed optical density can be assumed to be the same for neutron and photon exposures, and thus does not influence the sensitivity ratios.

Thermal neutrons. Table 7 gives the numerical values available for the estimate of the thermal-neutron sensitivity from equation (2). The silver and bromine concentrations were computed from the data of Table 1(a). Experimental data on the electron flux density required for a given photographic density ($\rho^{0.1}$) were available only for the film type 502.[19] The capture cross-sections and β-ray energies used are averages obtained from the values given in Tables 2(a) and (b). Table 8 shows the experimental results, as well as the results of the calculations. It is unfortunate that no better experimental data were available on film type 502, which is the only one for which an absolute estimate could be made. Even so, the measured and computed values are not out of line.

A comparison between Tables 6(b) and 8(b), giving the relative γ-ray-to-neutron or X-ray-to-neutron sensitivities of the three film types both for fast and for thermal neutrons, may be of interest. First of all, when neutron exposure is expressed in terms of flux, the ratio of neutron-to-γ-ray sensitivity of film type 510 is similar for fast and for thermal neutrons, a fact that is not easily understood. Also, it will be noted that while the ratio of sensitivities of the three film types to fast neutrons and to gamma rays is roughly proportional to the reciprocal of the average linear dimensions of the emulsion grains, all three film types show about the same ratio of thermal-neutron-to-γ-ray sensitivities. This is not too surprising, since the photographic response both of the γ-rays and of the thermal neutrons is due to the action of electrons of the same order of linear energy transfer, while the fast-neutron response is due to the action of protons. However, since, in the case of thermal-neutron exposures, the films never were exposed separately, an influence on the two adjacent films in the packet of the electrons produced by thermal neutrons in the relatively neutron-sensitive film type 510 cannot be ruled out.

4.3. *Variation of curve shape with processing conditions*

While for the experiments described so far all films were processed in X-ray developer, it may be of interest to refer here also to another experiment, for which a less vigorous developer was employed. Films of the same type and batch as those exposed to 3-MeV neutrons at NBS and to the thermal neutrons at ORNL, were also exposed to the fission flux from the LASL bare critical assembly (Godiva I).*

* All exposures and dose determinations were carried out by the staff of the Biology Division of LASL.

Table 8

(a) *Thermal neutron flux required for a given optical density*

Film type	Thermal neutron flux (n/cm²) for			
	$D = 0.05$		$D = 0.10$	
	Calculated	Measured	Calculated	Measured
502	—	—	$4.5\rho \times 10^9 = 2.5 \times 10^8$	$\sim 7 \times 10^8$*
510	$1.1\rho \times 10^9$	1.1×10^{10}†	$2.2\rho \times 10^9$	2.0×10^{10}†
606	$6.0\rho \times 10^9$	2.5×10^{10}†	$1.2\rho \times 10^{10}$	4.5×10^{10}†

(b) *Ratio, thermal-neutron-to-γ-ray sensitivity*

Film type	Reciprocal of measured thermal-neutron flux (cm²/n) required to produce the same density as 1 r of γ- or X-radiation	
	from $D = 0.05$	from $D = 0.10$
502	—	$\sim 1/(5 \times 10^9)$
510	$1/(4.2 \times 10^9)$	$1/(4.1 \times 10^9)$
606	$1/(3.6 \times 10^9)$	$1/(3.7 \times 10^9)$

* ORNL data.
† LASL data.

These films, together with a set of films of the same batch exposed to X-radiation from a 10-MeV betatron, were developed simultaneously in Kodak D-76 developer. Fig. 8 shows the results of this work for two of the three film types, together with a curve obtained on one of the film types exposed to X-rays from the 10-MeV betatron but developed in Kodak liquid X-ray developer. The photographic densities obtained with X- and γ-ray exposures are plotted against exposure in roentgens; the neutron data are plotted against first-collision dose rather than against flux, because the exposure data were reported to us in these units by LASL.

The X- and γ-ray curves obtained after processing in X-ray developer are linear on the log-log scale, just as the curves shown in Figs. 2 through 7. A similar linear response was obtained with the same film type by MURPHY for films exposed to a fission flux from Godiva II and developed in X-ray developer.[20] On the other hand, the curves obtained after processing in D-76 show a considerable difference in slope for different photographic densities; for any given density, the slope of the X-ray curve is different from that of the neutron curve. This difference would have been even more pronounced if a correction had been made on the neutron curve for the γ-contamination of the neutron beam employed; this contamination was of the order of 20 per cent. On the basis of these findings, it seems that it is impossible to assess quantitatively the influence of neutrons on the accuracy of photographic X- or γ-ray dosimetry when the film material is processed in a weak developer, such as Kodak D-76.

5. SUMMARY

The response of commercial photographic films to 3-MeV neutrons, which is to a large extent due to recoil protons, is found to depend

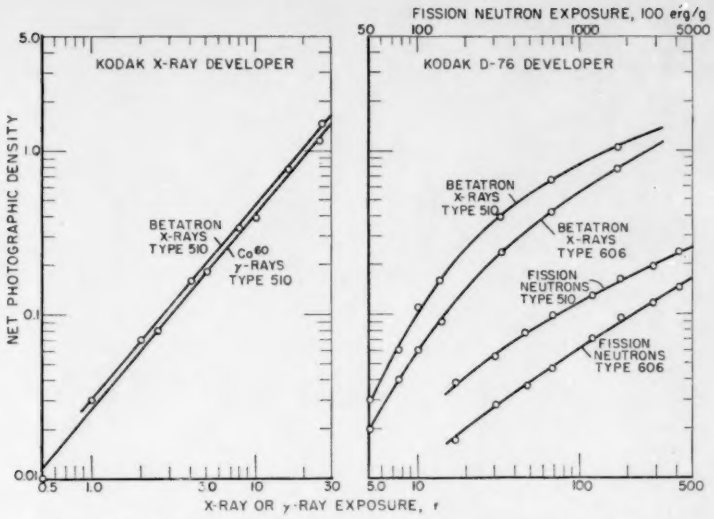

FIG. 8. Variation of shape of density vs. exposure curves with processing conditions.

on the amount of hydrogen in the emulsions and their surroundings, on the hydrogen-to-silver-halide ratio, and on the size and concentration of the silver halide grains. For a given hydrogen-to-silver-halide ratio, the response decreases with the number of grains encountered along the proton paths and with the fraction of these grains that is actually made developable. No experimental information is as yet available on the energy dependence of the fast-neutron response of film material. It is planned to obtain this information indirectly, by studying the energy dependence of the proton response of the same film types.

A comparison of the fast-neutron and the X- or γ-ray sensitivities of the three film types shows that—while, qualitatively speaking, fast-neutron sensitivity increases with increasing photon sensitivity—quantitatively, a considerable difference exists between the sensitivity ratios of the individual film types. Because of the waste of proton energy on large sensitive grains, film similar to DuPont film type 510, which has a high concentration of small grains of high sensitivity, is expected to be relatively more sensitive to fast neutrons than film with a small number of larger grains of equal sensitivity. In the three films studied, the fast-neutron sensitivity and the X- or γ-ray sensitivity were roughly in the ratio of the average reciprocal grain diameters of these films.

The photographic effect of thermal neutrons is largely due to the electrons released in the capture of thermal neutrons in silver and bromine. In terms of flux, the thermal-neutron sensitivity was found to be of the same order as the 3-MeV neutron sensitivity of these films, and essentially paralleled their X- and γ-ray sensitivity; i.e. the ratio of the thermal-neutron-to-photon sensitivities was not found to vary considerably with film type, at least when the films were in contact with each other during exposure.

While it is difficult to make good theoretical estimates of the neutron sensitivity of photographic films, the rather sketchy estimates so far obtained seem to indicate that, for a rough order-of-magnitude calculation of a film's fast- and thermal-neutron sensitivities, tedious experimentation may be dispensed with, provided that reliable information is available on the same film's proton and electron sensitivities.

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SHIELDING CALCULATIONS FOR CIVIL DEFENSE*

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Abstract—This paper discusses how the γ-ray penetration theory developed at the National Bureau of Standards has been applied to fallout shielding problems. The concepts of barrier shielding and geometry shielding are discussed and some comparison with experiment is given. The proposed publication of results is also described.

For many years the National Bureau of Standards has been engaged in a program to study the basic penetration of nuclear radiations. Most of the Bureau's calculations employ the moments method developed by Spencer and Fano[1]. Application of this method to γ-ray penetration resulted in an extensive publication of build-up factors by Nuclear Development Associates. Although the concept of build-up factor represented a great advance in penetration theory the results were still restricted to penetration in infinite media.

In order to overcome this restriction the National Bureau of Standards agreed to undertake a program in co-operation with the Office of Civil and Defense Mobilization to calculate angular distributions of γ-radiation. Knowledge of the angular distributions was expected to provide answers to the many problems involving finite shields. For the past three years the Bureau has been developing this program and also co-operating in many γ-ray shielding experiments in an effort to check theory with experiment.

The general problem considered in these calculations involves a source, a finite barrier and an isotropic point detector, as shown in Fig. 1. In general we separate the problem into one of *barrier* shielding and one of *geometry* shielding. Barrier shielding refers to the protection afforded by the barrier alone. It is calculated for a detector immediately adjacent to the barrier. Geometry shielding refers to the additional protection gained by moving from the position of detector A in Fig. 1 to the position of detector B.

We consider first barrier shielding by calculating the dose at detector A adjacent to the barrier. To solve this problem we must specify the *energy* and *angular* distribution of the incident radiation. The problem has been programmed for an IBM 704 computer in a general enough way so that an arbitrary energy spectrum of incident radiation may be considered. Since the Office of Civil and Defense Mobilization was primarily interested in the penetration of fallout γ-radiation, many of the first calculated results correspond to the spectrum from fission products at 1 hr after fission.

The computer program is also general enough to handle an arbitrary angular distribution of

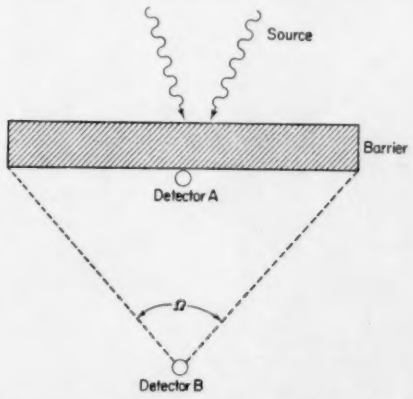

Fig. 1. General shielding problem consisting of γ-ray source, barrier, and γ-ray detector.

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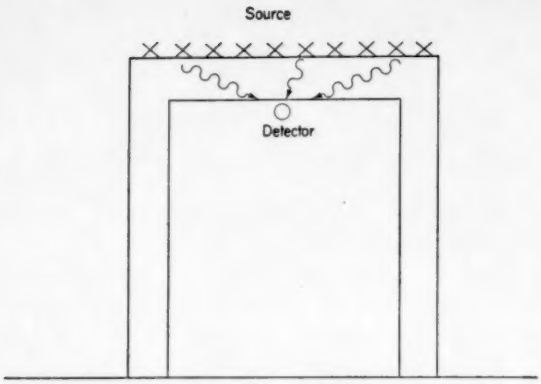

FIG. 2. Plane isotropic source of radiation generated by fallout uniformly distributed on a roof.

incident radiation. The angular distributions which were studied were determined by the geometry of the fallout situation. One obvious choice was the angular distribution from a plane isotropic source, which corresponds to the case of fallout distributed on a roof (see Fig. 2).

An experimental check on the calculations for the plane isotropic source of fission product gamma rays was provided by TITUS[2] who measured the penetration of actual fallout radiation through concrete slabs of varying thickness. The results of his measurements are shown in Fig. 3. This figure shows the relative dose rate as a function of depth in concrete for a plane isotropic source. The experimental points were measured during two shots at Operation Plumbob. The solid curve shows the attenuation calculated for the spectrum from fission products at 1 hr after fission. Since the data have been arbitrarily normalized, only the relative slopes are of interest. It can be seen that the experimental values agree very well with the penetration calculated for the fission product spectrum. The calculated attenuation curves for Co^{60} (1.25 MeV) and 0.7 MeV source energies, shown by dashed curves, have been included for comparison. Although 0.7 MeV is often used as an effective source energy of fallout radiation, the experimentally determined slope at 1 hr after fission is better approximated by a source energy nearer to that of Co^{60}.

Additional protection is given by geometry shielding as the detector is moved away from

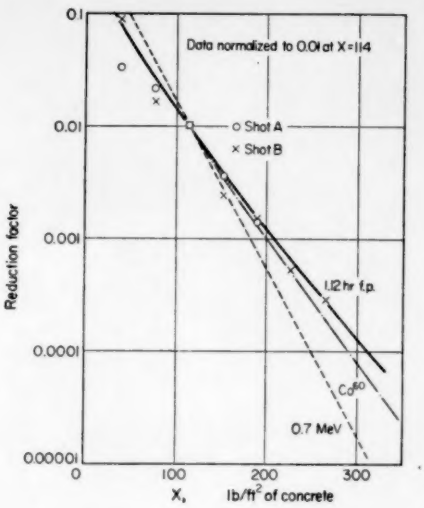

FIG. 3. Comparison of experimental and calculated values of barrier shielding from a plane isotropic source of fallout radiation.

the barrier (detector B, Fig. 1). As the solid angle Ω which the barrier subtends at the detector is decreased, the dose rate, of course, goes down. Just how much the dose rate decreases will depend on the angular distribution of the energy flux at the detector. Calculations show that the angular distribution at a given point behind a barrier depends on the barrier thickness. Since the geometry factor is obtained from an integration over the angular distribution, it too will depend on barrier thickness. The geometry factor $G(\Omega, x)$ behind a barrier of fixed thickness is given by

$$G(\Omega, x) = \frac{1}{I_0} \int_{\Omega} I(\omega, x)\, d\omega$$

where the integral is taken over all directions ω contained within the solid angle Ω subtended by the barrier. In this equation x is the barrier thickness, $I(\omega, x)$ is the angular distribution of the energy flux, and

$$I_0 = \int_{2\pi} I(\omega, x)\, d\omega$$

is the integral of the angular distribution over the forward hemisphere ($\Omega = 2\pi$).

The dependence of the geometry factor on the barrier thickness implies that the rate at which

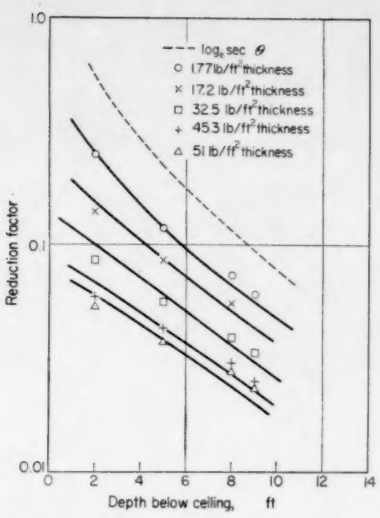

FIG. 4. Experimental and calculated values of shielding beneath the roof of a blockhouse, illustrating the dependence of geometry shielding on barrier thickness.

the dose rate decreases as a function of distance behind the barrier will vary with barrier thickness. Confirmation of this effect comes from experiments at the Army Chemical Center in Edgewood, Maryland, by MORGENTHAU and REXROAD[3]. They have measured the dose rates at various depths below the roof of a concrete blockhouse for several different roof thicknesses. Their experimental results are plotted in Fig. 4 as a function of the depth below the ceiling of the blockhouse for various roof thicknesses. The data are presented in terms of the reduction factor, which is defined as the ratio of the dose rate at the detector to that at 3 ft above a smooth infinite plane of the same source density. The solid curves are values calculated at the National Bureau of Standards for the corresponding roof thicknesses. The dashed curve is the value calculated for zero roof thickness. It can be seen that the experimental data verify the fact that the rate of decrease depends on the barrier thickness. Since the variation of the reduction factor with depth is less for greater roof thicknesses, the conclusion is that the angular distribution is more sharply peaked in the downward direction for the greater roof thicknesses.

The penetration of radiation from sources on the ground through the walls of a building has also been treated by a combination of barrier and geometry shielding. First the barrier shielding was calculated for the fission product energy spectrum and for the angular distribution of the radiation incident on a vertical wall in a fallout field (see Fig. 5). Calculated results give

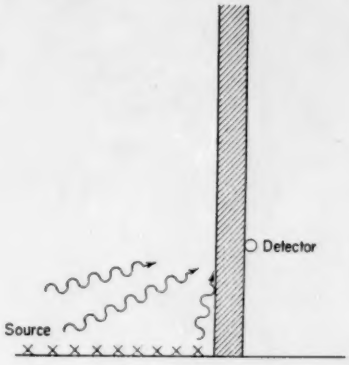

FIG. 5. Radiation incident on a vertical wall in a fallout field.

the attenuation of the wall for a semi-infinite source on one side and a detector on the other. In order to discuss geometry shielding for radiation from the walls, it is convenient to consider the detector at the center of a four-walled building, as shown in Fig. 6. In addition to the two solid angles shown, the solid angles subtended by the front and rear walls also had to be calculated. The geometry factor for each

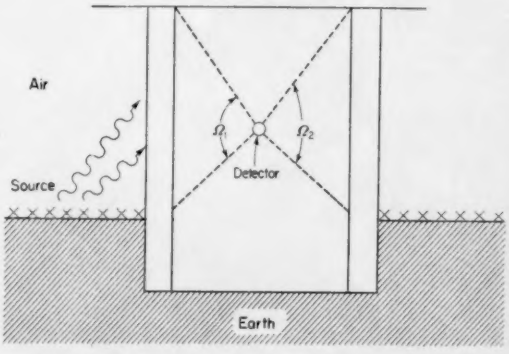

FIG. 6. Solid angles subtended at a detector by the walls of a building. Building is surrounded by an infinite fallout field.

wall was calculated and the results added to obtain the geometry factor for all walls.

Basic calculations from this program will be published by the National Bureau of Standards as they are completed. The calculations for the fallout energy spectrum and their application will be incorporated in three shielding manuals, soon to be published. The first manual, by L. V. SPENCER, describes in detail the results of the machine computations and their application to shielding problems. A second Engineering Manual to be published by OCDM, applies SPENCER's results to the design and analysis of structures for protecting against fallout radiation. The third manual[4] gives simplified calculations to determine protection when one is interested in a survey of a large number of structures.

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THE RADIATION HAZARD FROM CONTAMINATED AIRCRAFT*

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Abstract—This report describes a study of the relative importance of the external, inhalation and ingestion hazards associated with surface contamination of aircraft that have penetrated atomic clouds or the stratosphere. Experiments were conducted at Operations Redwing and Plumbbob. They included γ- and β–γ-aircraft surveys, air sampling, swipe sampling, radiochemical and biological analyses and personnel dosimetry.

The latest maximum permissible total radiation dose limits and the NCRP total organ burdens are interpreted in terms of the aircraft maintenance problem. Curves are drawn to show the maximum permissible concentration of mixed fission products in air as a function of age of the debris assuming a 40-hr work week for both controlled and uncontrolled situations. Similar curves are given for the ingestion hazard.

The radiological hazard from external radiation is compared with that presented by inhalation and ingestion. It is found for mixed fission debris on aircraft ranging in age from 1 hr to 1 year that the external radiation hazard is dominant by large factors under all normal working conditions. It is concluded that γ-surveying is generally adequate to define the radiation problem. Release to uncontrolled areas appears warranted if the average measured γ-field in potential working areas around the aircraft or its parts is less than 0.5 mr/hr regardless of age. Swipe sampling cannot be used to evaluate the local air concentration and thus the inhalation hazard. For mixed fission products on aircraft surfaces swipe sampling is no more reliable than a close β-survey of the surface for estimating the potential ingestion hazard.

INTRODUCTION

EACH radiation health problem requires the interpretation of the basic rules of permissible dosage set by the National Committee on Radiation Protection. The situation to which this study is addressed is the radiation hazard to aircraft maintenance personnel from aircraft contaminated with mixed fission products. The contamination is produced by flight through fresh atomic clouds generated at test sites or older debris in the stratosphere. The experimental investigation has attempted to procure the necessary information to make it possible to define the relative contribution of external radiation, inhalation and ingestion to the total hazard, and to assess the absolute levels expected for various types of flight conditions. Once these are defined it becomes possible to establish operational procedures which can insure safe and efficient operation.

The common problem is that of servicing aircraft shortly after they have passed through an atomic cloud. For this situation radiation control will almost always be exercised so that relatively high levels are acceptable. Some weeks later some of these aircraft or their parts may be returned to maintenance depots where they will be handled. Normally it is possible to set aside an engine from such an aircraft for periods up to 6 months without serious inconvenience so that considerable reduction in activity can be obtained with little effort. Finally, certain aircraft that fly consistently above the tropopause may become slightly contaminated with the relatively old mixed fission debris

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that comprises the stratospheric reservoir. This may become significant for commercial jet aircraft in the winter in the northern latitudes where the polar tropopause may descend to 30,000 ft or less.[1] The period of interest is therefore from a few hours where γ-survey and strict control are effective, such as at the test site, to low levels of contamination with mixed fission products that may be nearly a year old (the average apparent age of the present stratospheric "background").

At the time this investigation was initiated there were large areas of uncertainty in the relative contribution of ingestion, inhalation and external radiation for workers on a contaminated aircraft. The distribution of the activity on the aircraft engine and external surfaces was not clearly defined. Almost nothing was known of the extent of transfer of the contaminant from the aircraft to the surrounding air or to men's hands. A preliminary evaluation of this problem based in part on the data presented here has already appeared.[2]

The experimental measurements were made on aircraft that had penetrated fresh atomic clouds produced by a variety of nuclear devices at Operation Redwing at Eniwetok in the summer of 1956 and Operation Plumbbob in Nevada in the summer of 1957. Samples were also taken of the stratospheric debris over the continental U.S. during periods of no nuclear testing. The use of fresh high-level contamination not only makes the measurements simpler and more rapid but provides information on the worst situations that may be encountered. In general it was found that there was no significant difference in the relative contributions of the three sources of radiation hazard regardless of the size of the burst, the chemical composition of the device or the height above land or water. Chemical fractionation in small ground-level bursts may alter the ratio of certain critical long-lived isotopes to the total β-activity, but even in the most extreme case this will not affect the radiation hazard problem on aircraft surfaces in any important way.

The measurements included: (a) complete aircraft surveys for relative and absolute activities of α-, β- and γ-radiation; (b) air sampling to relate the concentration of the radionuclides in the air to γ-fields and levels of surface contamination; (c) swipe tests utilizing various chemicals and techniques of swiping; (d) efficiency of removal of surface contamination with the hands; (e) transfer of the radioactive contaminant from the hands to the mouth via bread or cigarettes; and (f) comparison of the hazard of work on the ramp with work in a hangar.

The focus of the field study at Operation Plumbbob was the maintenance man who was followed closely during the entire operation so that all of his radiation dose from contaminated aircraft could be understood. The measurements included; (a) the γ-field he worked in; (b) the β-radiation of the surfaces he touched; and (c) the radioactivity of the air he breathed. In addition, urine samples taken just before and after the operation provided a monitor of body buildup of long-lived fission products.

THEORETICAL CONSIDERATIONS

During the past few years there has been a rapid increase in the biochemical and radiological data applicable to health physics with the result that many standards have been revised or redefined. The biological half-lives, fraction of isotope reaching the critical organ and the maximum permissible internal concentration of radioisotopes were taken from the new compilation by the Sub-Committee on Permissible Internal Dose of the National Committee on Radiation Protection published in Handbook 69.[3] The physical decay constants are taken from STROMINGER *et al.*[4] It is recognized that the retentivity of radionuclides in the body can be described more accurately by a power function than a simple exponential,[5] but in the absence of these data on most of the nuclides and the approximate nature of the calculations to be made, the biological half-life will be used. The composition of the mixed fission debris for U^{235}, Pu^{239} and U^{238} has been published.[6] Variations in the composition of the critical long-lived isotopes as a result of thermal fission of Pu^{239} or fast fission of U^{238} will not exceed a factor of 2.

Since the contaminant consists of mixed fission products, a substantial amount of information is available on its physical, chemical

and isotopic character as a function of age, type and height of burst and fissionable material. On the other hand, there is a considerable inherent uncertainty in the isotopic composition of young clouds (at least a factor of 2 or 3 in such critical ratios of I^{131}/Sr^{90}) as a result of these factors. This is probably no larger than the uncertainty in the biological effects.

Age of the contaminant

Since, as will be seen below, the acceptable measured dose varies strongly with the age of the mixed fission products, a simple method is needed to determine this parameter. Isotopic analysis is difficult and slow and requires special laboratory facilities. Ingenious methods have been proposed which characterize the radiation in a general way and thus obtain a crude estimate of the age in one measurement.[7] For practical operation in the field or in the repair shop it still seems best simply to make two measurements of the radiation field over a suitable time interval. This may be done by taking a swipe sample, mounting on a planchet and following the decay. It can also be done with any reliable γ-survey meter if the levels are high enough and the debris is only days to weeks old.

It has been shown that the decay of the mixed fission products can be approximated with sufficient accuracy for present purposes by the relation

$$A = A_0\left(\frac{t}{t_0}\right)^{-1.2}$$

where t is the time since detonation and A_0 is the initial intensity at t_0 (t must be measured in units of t_0). Thus, if A_1 and A_2 are the activities of a sample at two different times, t_1 and t_2, and $F = A_2/A_1$, $\Delta_t = t_2 - t_1$

$$F = \left(\frac{t_1}{t_2}\right)^{1.2} = \left(\frac{t_1}{t_1 + \Delta_t}\right)^{1.2}$$

and rearranging

$$t_1 = \left(\frac{F^{0.833}}{1 - F^{0.833}}\right)\Delta_t$$

The last relation gives the age of the contaminant at the time of the first measurement. A nomograph (Fig. 1) permits rapid evaluation of such measurements and is useful in estimating the time interval required for an age determination of a given precision. For practical purposes it is only necessary to determine whether the age of the debris is 1, 10 or 100 days in order to permit a preliminary evaluation of the levels suitable for controlled or uncontrolled servicing of the aircraft.

Health physics standards

The most recent statement of the National Committee on Radiation Protection, 1958,[8] for controlled areas recommends that the maximum permissible accumulated dose in rems for an individual of any age, is equal to five times the number of years beyond age 18, but never at a rate in excess of 12 rems per year. This applies to the whole body and to all critical organs except the skin for which the permissible dose in rems is ten times the number of years beyond age 18 under operational conditions. The 13-week dose is 3 rems for the whole body, 6 rems for total skin and 25 rems for hands and forearms only. Finally, a dose of 0.5 rem per year is accepted for permissible whole-body exposure for uncontrolled areas.

For internal exposure the maximum permissible concentration in any particular organ of all isotopes contributing radiation to that organ is 3 rems per 13 weeks, unless the critical organ is the gonads or the whole body in which case no more than 1 rem per 13 weeks is recommended.

In the case of inhalation the maximum permissible concentration in air normally assumes continuous breathing of the contaminated atmosphere 168 hr per week. The maximum permissible concentration in water is set with the assumption that all water drunk by an individual would be contaminated at this level.

When in equilibrium with the associated corpuscular emission, ordinary X-rays (e.g. 200 keV) produce a local energy absorption of approximately 93 ergs/g of tissue per r. For soft tissue this value is not strongly dependent on the energy of the X-rays in this region. Therefore, in this special but useful case, the roentgen essentially equals the rad. Further, since

$$\text{dose (in rems)} = \text{dose (in rads)} \times \text{RBE}$$

where RBE is the relative biological effectiveness and the RBE for ordinary X-rays is

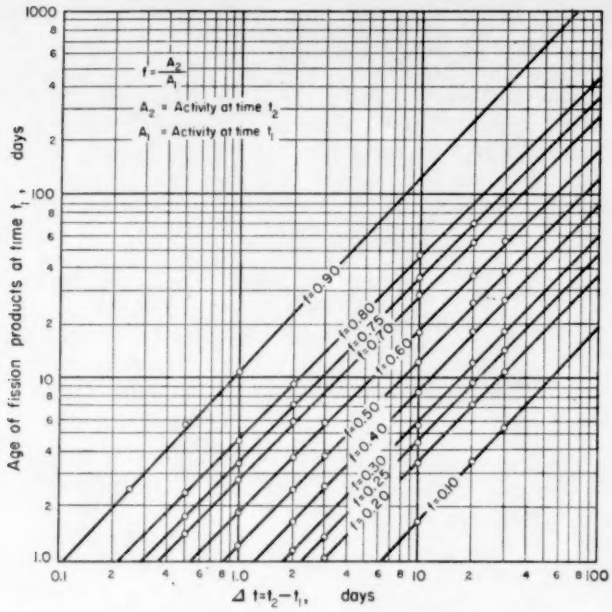

FIG. 1. Age determination of fission product contamination.

taken as unity, the measured dose in roentgens approximately equals the absorbed dose in rems if there were no self-shielding and geometry factors. The average γ-energy of mixed fission products is reasonably constant at about 0.7 after a few days. Therefore, the measured roentgen value in a properly calibrated survey meter will give an adequate estimate of the effective dose in the body in rems from the radioactive contaminant on the aircraft (again neglecting self-shielding and geometry). Fig 2. shows that in this energy range even the dose to the bone does not depart significantly from that predicted from the measured ionization in air (roentgen).

In general, as will be shown below, the β-activity will be of no consequence. On the average, the workman will be at least 1 ft away from the aircraft, half or more of his body will generally get no β-dose at all, and his clothing will reduce β-dose just at the skin surface to something less than the γ-dose. Therefore, except when using bare hands or in unusual situations the β-contribution can be neglected for the external radiation exposure.

Dose from external sources. In a radiation control area such as occurs at the Nevada and Pacific Proving Grounds, the permissible level is clearly defined. The men must remain below 3 rems/13 weeks. Assuming a 40-hr work week (actual direct work on aircraft is generally much less than this due to setup time, errands, parts requirements, coffee breaks, etc.), operation could be continuous for weeks at a time in a radiation field which provided an average dose of about 6 mr/hr. For continuous exposure this would have to be reduced to an average dose rate of 2.5 mr/hr to satisfy the over-all requirement of 5 rems per year. Much higher levels would be acceptable, of course, for shorter, i.e. actual, work times, providing the 13-week average is met.

For continuous 40-hr work week in uncontrolled areas, i.e. general population, the permissible level is a factor of 10 lower, i.e. a dose rate of 0.25 mr/hr. A maintenance depot that had a continuous flow of lightly contaminated aircraft at this level could maintain operations on an uncontrolled basis. Due to geometry and self-shielding of the body, measurements made in this study have shown that the actual dose for persons working on contaminated aircraft

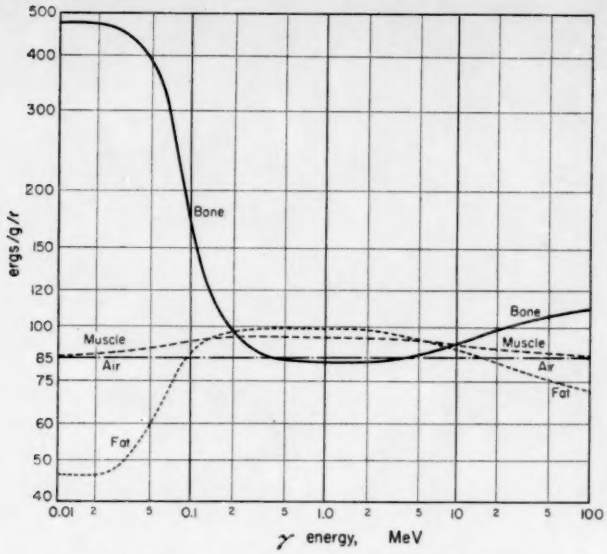

FIG. 2. γ-Absorption in various tissues (from O. GLASSER (Editor), *Medical Physics*, Vol. II, p. 791, 1950. By permission of Year Book Publishers, Inc., Chicago).

is generally about one-half of the meter reading of a properly calibrated γ-survey instrument employed at waist height. Thus the measured dose rate of 0.5 mr/hr is the highest that is acceptable for any position of continuous work on an aircraft in an uncontrolled area where the occupancy cannot exceed 40 hr per week. If all such positions are less than this value, the aircraft may be released to uncontrolled areas. For the reasons stated above, this is very conservative since no workman will stay at one position for 40 hr per week.

Higher levels could be acceptable in an uncontrolled area (no film badges or individual dosimeters) if a technical person simply measured the γ-field around an aircraft on arrival at the depot. By knowing the number of such aircraft in the shop, the age of the debris, the history of the shop and the level on each aircraft, they could be programmed in such a way that even on an uncontrolled basis the workmen would not exceed 0.5 rem per year even though measured fields considerably above 0.5 mr/hr were present for short intervals.

In the rare event that a single aircraft contaminated with mixed fission products were sent to an otherwise uncontaminated maintenance depot that would normally not receive contaminated aircraft, it would readily be possible to set the maximum permissible γ-field as a function of the age of the contaminant arrival. A series of curves can be drawn which could stipulate the acceptable level providing the age of the debris and the time the aircraft will be in the shop are known. In the next more complicated case of intermittent arrival of contaminated aircraft separated by several weeks, it would be possible for a health physicist to program the levels such that the workmen could not possibly exceed 0.5 rem per year. This approach would not require instituting radiation control with attendant costs. Clearly, if a steady flow of aircraft were to be serviced at levels of about 0.5 mr/hr or higher, radiation control should be instituted. This in turn would permit the permissible level to be raised by a factor of 10. Here the actual level on the aircraft would only be important if it were to reduce the work hours of critical personnel below that required for operational efficiency. In most cases simple storage to allow the activity to decay would be the preferred solution.

The fundamental relationship in making all such computations as suggested above is

$$D = \int_{t_1}^{t_2} A_0\, t^{-1.2}\, dt = 5\, A_0(t_1^{-0.2} - t_2^{-0.2})$$

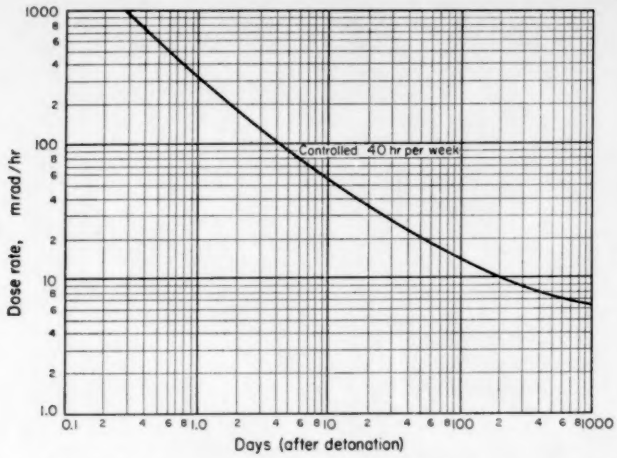

FIG. 3. Maximum permissible γ-levels for whole body vs. fission product debris.

and since $A_0 = A_1 t_1^{1.2}$ for mixed fission products

$$D = 5\, A_1\, t_1^{1.2}(t_1^{-0.2} - t_2^{-0.2})$$

where D is the total external dose received during the time interval $t_2 - t_1$, A_0 is the activity at some earlier time t_0 (t_1 and t_2 are measured in units of t_0) and A_1 is the activity at time t_1. Thus, by rearranging, the activity or initial dose rate, A_1, from mixed fission products, can be stated which will give any desired dose D in the time interval $t_1 - t_2$, i.e.

$$A_1 = \frac{D}{5\, t_1^{1.2}(t_1^{-0.2} - t_2^{-0.2})}$$

It is clear that the acceptable γ-dose from a source of mixed fission products at any given time is a function of the age of the debris. The younger the debris, the smaller the integrated dose for a given dose rate. Fig. 3 shows curves of dose rate vs. age of the fission debris to keep the integrated dose below 3 rems per 13 weeks for a controlled 40-hr work week.

Inhalation. The maximum permissible concentration of a given isotope in air is the concentration of that isotope which produces an equilibrium concentration in some specific body tissue such that a dose of 3 rems/13 weeks is given. (In the case of the gonads, or whole body, this level is reduced to 1 rem/13 weeks.) To compute these MPC's it is necessary to know the effective half-life of the isotope in the organ, the effective energy of the radiation, the distribution of the isotope in the body and the fraction which enters the lungs which ultimately gets to the critical organ. A full discussion of the derivation of the equations and the definition of terms is given elsewhere.[9] Using new MPC values for the sixty-two most important fission product isotopes,[3] and for eight more isotopes calculated by them, SUMMERS *et al.*[9] obtained MPA's (maximum permissible concentration in air) for twelve different organs as a function of the age of the fission products. The lowest MPA for any of the critical organs at each age after detonation was chosen as the MPA of the mixture at that time. Their results give the following critical organs and times: GI tract (insoluble), from 15 min to 7.8 days; thyroid 2.8–3.9 days; lungs (insoluble) 3.9–38.5 days; and bone from 38.5 days to 81 years. It is known that virtually all of the strontium–rubidium, cesium, barium and iodine, and a considerable fraction of the other isotopes are soluble in water or dilute hydrochloric acid. If the same calculation is made assuming these solubilities (instead of assuming that these isotopes are insoluble as was done by SUMMERS *et al.*[9]) but using only the ten most important isotopes, the thyroid is the critical organ from a few hours to 14 days after which Sr^{90} in bone becomes dominant. The MPA curve calculated on this basis does not differ

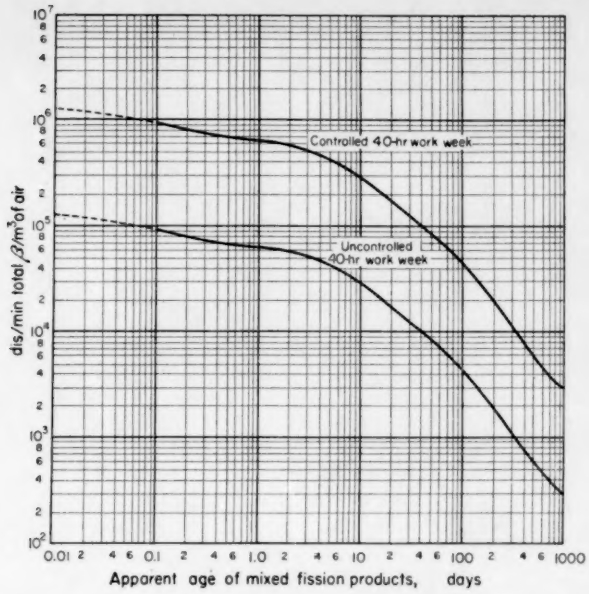
Controlled 40-hr work week
Uncontrolled 40-hr work week
dis/min total β/m³ of air
Apparent age of mixed fission products, days

FIG. 4. Permissible continuous inhalation levels.

from that of SUMMERS *et al.*[9] in any important way for debris older than 1 day. For younger debris the assumption that the debris is all insoluble leads to a lower MPA, e.g. at 1 hr this curve is a factor of 10 lower than that assuming complete solubility.

The curve given by SUMMERS *et al.*[9] is the MPA vs. age of mixed fission products for continuous breathing in an uncontrolled area. Fig. 4 shows the analogous curves derived from that of SUMMERS *et al.*[9] for the normal 40-hr work week in a controlled area. Also given is the curve which is applicable to a 40-hr work (i.e. occupancy) week in an uncontrolled area. This is the normal situation for maintenance personnel at depots or aircraft plants.

It is seen from Fig. 3 that the tolerable levels of mixed fission products in air are severely dependent on the age of the debris. Without taking cognizance of this fact wholly unrealistic permissible operating levels might be set with resultant expense and confusion.

The curve for non-controlled areas (Fig. 3) is computed on the basis of a 40-hr occupancy week since it is unlikely that even without control an individual could consistently stay next to any piece of equipment for a greater period per week. The curve shows that ten times as much activity can be accepted from 10-day old debris as from 100-day old debris and that another factor of 2 can be gained for 1-day-old debris.

Ingestion. The ingestion of radioactive debris from an aircraft is primarily a result of contamination of the hands during work time and subsequent transfer to the mouth via food. The problem may then be considered as a single dose daily for 5 days per week. It is required to determine the maximum permissible total β-activity that may be ingested per day. This can be done by first evaluating the MPC for individual isotopes and organs and then summing the effects in the most critical organ for fission products of various ages. SUMMERS *et al.*[9] have done this calculation for the maximum permissible total β-dose in drinking water (MPW), using seventy isotopes. Their results are directly applicable to the present problem since they assumed no fractionation, and if the GI tract were the critical organ they assumed all of the debris insoluble. They have produced a curve for the uncontrolled maximum permissible concentration of mixed fission products in dis/min per m³ against the age of the debris. Since standard man drinks 2200 ml per day but since

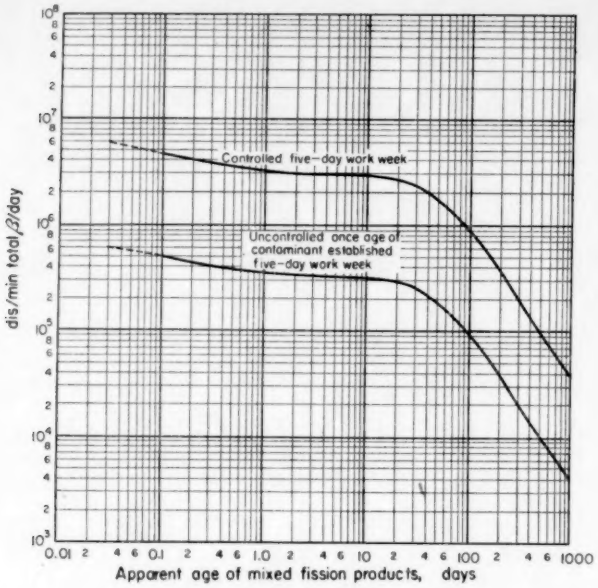
dis/min total β/day
Controlled five-day work week
Uncontrolled once age of contaminant established five-day work week
Apparent age of mixed fission products, days

FIG. 5. Permissible continuous ingestion levels.

we are concerned only with ingestion during working days, the maximum permissible ingestion MPI (in dis/min of total β-dose per day), for mixed fission products at time t after detonation for an uncontrolled work area can be obtained from the curve of SUMMERS *et al.* by the simple relation

$$(\text{MPI})_t = \text{MPW}_t \text{ (in dis/min per m}^3\text{)} \times 2.2 \times 10^{-3} \times \frac{7}{5}$$

This is plotted together with the curve for a five-day work week in a controlled area in Fig. 5. The shape of the curves is similar to those obtained for the inhalation problem and shows again the strong dependence on the age of the debris. It has been shown from the AEC balloon program[10] and other studies that the average apparent age (i.e. Sr^{90}/total β-dose) of stratospheric fission products does not exceed about a year at any point where aircraft may fly.

Working period. If all three sources of radiation dose are operative in a controlled area the allowable work week may be approximately derived as follows:

$$F = \frac{1}{F_e + F_i + F_g}$$

where F = fraction of the 40-hr work week that is permissible considering all sources of radiation

$F_e = I/300$, which is the fraction of maximum permissible dose from external γ-radiation that is obtained, I is in mr/40 hr.

$F_i = A/\text{MPA}$, which is the fraction of maximum permissible dose from inhalation where MPA is the maximum permissible total β-dose in air as discussed above and A is the measured total β-dose per unit value of air.

$F_g = G/\text{MPI}$, which is the fraction of the maximum permissible dose from ingestion where MPI is the maximum permissible daily total β-dose that may be ingested and G is the actual ingestion per day

(actually, this fraction is quantized by the day but, for the approximate solution, this will be neglected).

Nature and process of contamination

The mixed fission products are attached to, or incorporated into, dust particles during the development of the cloud. For high air bursts, surface water shots or very large explosions, there is very little isotopic fractionation due to close-in fallout of large particles. Further, the critical isotopes Sr^{90}, I^{131} and I^{133} are generally in soluble form. For surface shots of small size the range in the Sr^{89}/Mo^{99} ratio may be a factor of 10 as a result of chemical fractionation and for underground shots as high as 200. The ratio of iodine to strontium isotopes is very much less affected due to the volatile nature of iodine or iodide and the inert gaseous precursors of the strontium isotopes. It is concluded, therefore, that fractionation does not play a significant role in this problem, i.e. the ratio of I^{131}/Sr^{90} and the ratio γ/Sr^{90} or, total β/Sr^{90}, does not vary by more than a factor of 2 due to fractionation except in very rare cases.

Isotopic ratios Cs^{137}/Sr^{90}, Ce^{144}/Sr^{90}, Ru^{103}/Sr^{90}, Zr^{95}/Sr^{90} and Sr^{89}/Sr^{90} and total β-decay curves were obtained from debris from eleven bursts representing a wide variety of devices. In no case did the ratios differ by more than a factor of 2 from what was expected by the nature of the fission process. The total range in the ratios for all samples did not exceed a factor of 5. The maximum differences are found among the small bursts as expected.

The problem of aircraft contamination from flights in the stratosphere involves only megaton shots in which the variation in isotopic composition is much less. This contamination normally has an average age of months. Here the Cs^{137}/Sr^{90} ratio is the most important one. Measurements by STEWART *et al.*[11] show that the Cs^{137}/Sr^{90} ratio in stratospheric debris is close to 1.8, the theoretical value for the fast fission of U^{238}.

It was found that for all shots the total β-decay followed the $t^{-1.2}$ law. There were small variations in the value of this exponent from one burst to another and it changes somewhat with time. These variations are of no significance for the aircraft contamination problem and, therefore, predictions may be made with sufficient accuracy using this law from all shots and times.

Alpha-activity was searched for in a number of cases but could not be detected within the experimental error with a counter that had a background of about 15 counts/hr. Considering the efficiency of the detector and the area covered, it is concluded that even if the upper limit of the counting error were assumed to be due to Pu^{239}, the Sr^{90} hazard would be at least ten times as great in all cases.

Most of the radioactivity from an air burst resides on particles near 0.9 μ in size but the maximum number of particles is less than 0.1 μ and many as small as 0.01 μ can be seen. For a stable aerosol to exist the particle size must be less than 1 μ so that for most atomic clouds and certainly in the stratosphere this gives the order of magnitude of the particle size.

The actual contamination appears to occur by impaction of these small particles on the aircraft surface. This is indicated by the fact that the concentration of the contaminant on any leading surface of an aircraft was independent of the nature of the surface of the collector, whether aluminum foil, gummed paper, or masking tape. The maximum activity occurs on those portions of the aircraft surface where the direction of air flow is most abruptly changed. In detail, on any particular surface, the highest concentrations are observed at any rough spots such as occur at rivet heads, screws, etc. Fig. 6 shows an autoradiograph from a contaminated aircraft surface to illustrate this point.

To evaluate any of the radiological hazards with reference to aircraft maintenance men, it is necessary to know in some detail the variation of the radiation flux at various positions on and in the aircraft. A number of aircraft that had penetrated a variety of atomic clouds both at Redwing and Plumbbob were surveyed at predetermined positions for β- and γ-radiation shortly after their return from the contaminating operation. In some cases surveys of the engines were also performed. Repeat surveys after several days showed no change in the relative activity on the different parts of the aircraft.

Table 1. *Normal variation of radiation level on contaminated aircraft*
(Relative values in mr/hr)

Position	Surface contamination β-field at 1 in.	γ-Field at 1 ft
Nose	1500	150
Leading edge of wing outboard	1000	200
Leading edge of wing inboard	1500	350
Nose of tip tank	1000	100
Side fuselage near wing	400*	350
Cockpit	10	100
Fuselage at rear	50	50
Leading edge horizontal stabilizer	400	80
Wheel and wheel well	10	350
Starter hub	2000	300
Top of wing outboard	100	100
Top of wing over engine	1000	300
Just inside tailpipe	3000	300
Bomb bay lip	1000	300
After engine demounted		
engine well in wing	200	200
exhaust stack in wing	10,000	1000
exhaust cone engine	10,000	1000
rotor and stator blades engine	10,000	1000
exterior engine surface	500	1000

* Film samples would indicate this should be 100 if there were no contribution from scattered γ-radiation.

The normal distribution of radioactivity on the surface of an aircraft that has flown through an atomic cloud can be inferred from the β-activity as shown in Table 1. This table gives the average result from twenty-one surveys of various aircraft contaminated during eleven different atomic explosions. The measurements on each plane were normalized to the leading edge of the wing outboard being taken as 1000 mrads/hr β-dose at 1 in. This distribution is essentially independent of the aircraft and the type of the burst. Such differences as are observed are probably due in part to variations in measurement technique, differential washing depending on the orientation of the plane on the ramp relative to rains, and the relative contribution to the apparent β-count from scattered soft γ-radiation that originates in the engine. This latter is very critical for measurements on the wing over or under the engine. In general, measurements on the same position on either side of a particular aircraft agree within a factor of 2.

The maximum range of activities on a single aircraft appears to be about a factor of 1000. The highest levels are observed on the exhaust cone and turbine blades of the engine which are about ten times the leading edge of the wing. Inside the cockpit or other reasonably sealed positions such as the wheels, the inside of the aileron hinge, the air-conditioning element, or the fuel manifold, the level is generally a factor of from 10 to 100 less than the leading edge of the wing. All of the leading edges—wing, nose, tip tank, or stabilizers—are similar in surface contamination level.

Also given in Table 1 are the estimated γ-fields to which a workman might be exposed if he were near the position indicated.

It may therefore be seen that the entire contamination pattern of an aircraft may be reasonably estimated from a survey of the leading edge of the wing or even the nose. The leading edge is preferable as it presents the possibility of a number of measurements whose average will be more reliable than the single measurement on

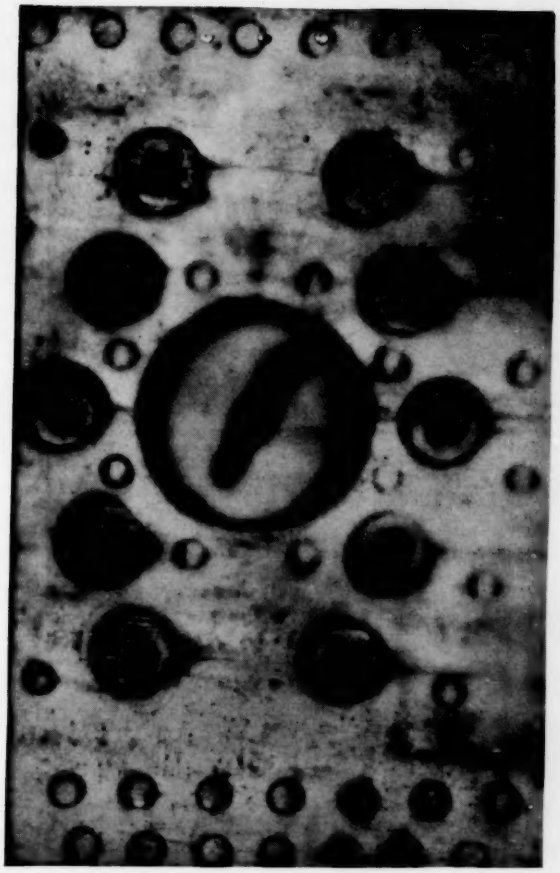

FIG. 6.

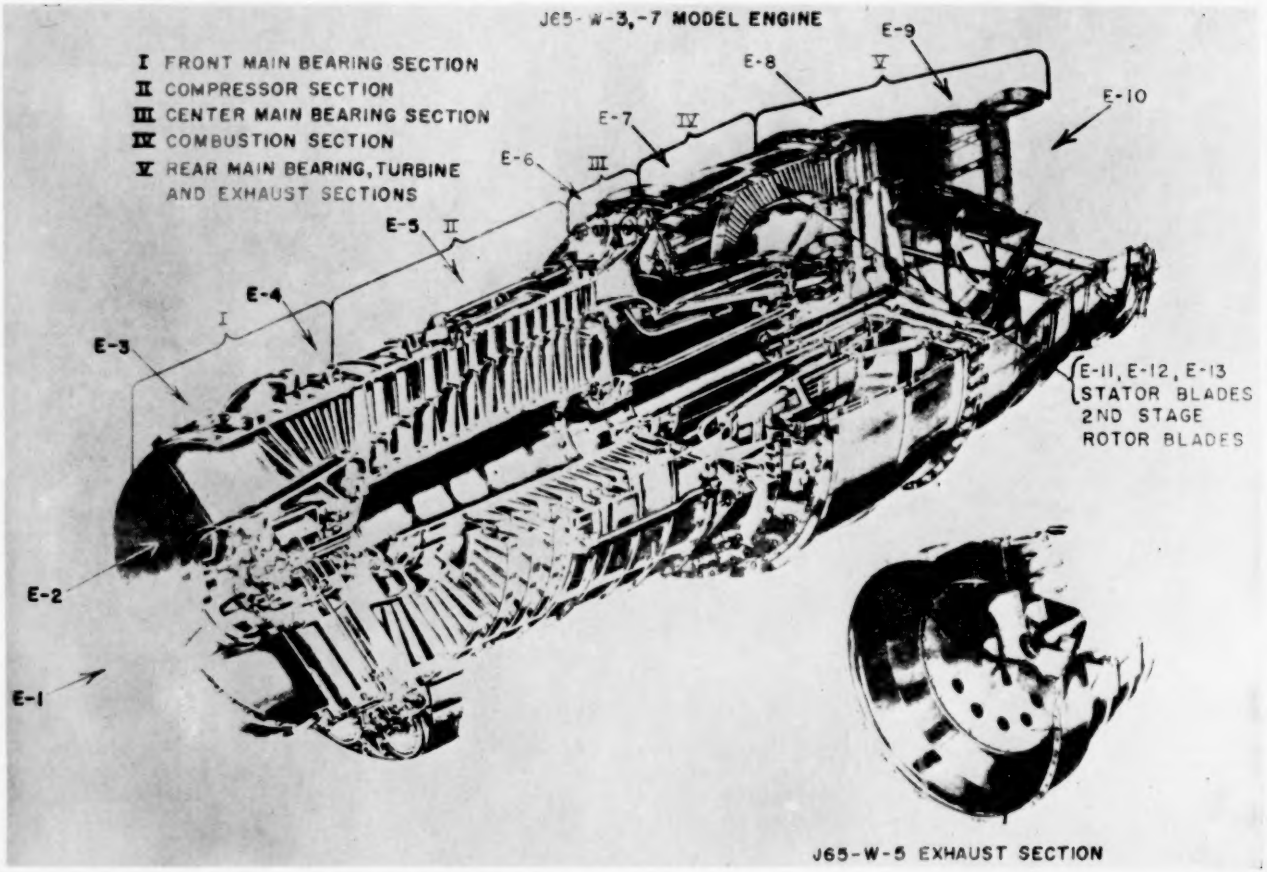
J65-W-3,-7 MODEL ENGINE
I FRONT MAIN BEARING SECTION
II COMPRESSOR SECTION
III CENTER MAIN BEARING SECTION
IV COMBUSTION SECTION
V REAR MAIN BEARING, TURBINE AND EXHAUST SECTIONS
E-1
E-2
E-3
E-4
E-5
E-6
E-7
E-8
E-9
E-10
E-11, E-12, E-13
STATOR BLADES
2ND STAGE
ROTOR BLADES
J65-W-5 EXHAUST SECTION

FIG. 7.

the nose. The total range of surface contamination subject to human handling varies over a range of at least 1000. This does not include cockpit surfaces in a plane with a tight filter system, nor all interior parts of the aircraft engine. These surfaces might extend the range, although the rotor and stator blades which were measured were probably as contaminated as any part of the engine. The γ-field near the aircraft varies only over a range of a factor of 10. The γ-flux near the unshielded engine, however, is higher than the highest flux on the exterior of the aircraft by a factor of 2 or 3.

External hazard

Aircraft flown deliberately through fresh atomic clouds (e. g. $H + 20$ min) are exposed to an integrated dose of 2–20 r during passage through the cloud. During the first few hours after returning to base the aircraft are normally allowed to "cool". At $H + 6$ hr the average γ-field 1–2 ft away from the leading edge near the fuselage is about 100 mr/hr. A day later this drops to about 12 mr/hr, after 4 days to 4 mr/hr, after 3 weeks to about 0.5 mr/hr and after 6 months to about 0.05 mr/hr. At all times, the highest level is directly over the exposed engine which remains at a factor of 10 higher than those for the area of the leading edge near the fuselage. Note, however, that after 6 months, even engines flown directly through a fresh atomic cloud are acceptable in an uncontrolled area.

A typical maintenance operation occurred during Operation Redwing in which eight men were required to overhaul the engines of a B57 aircraft for a 50-hr inspection. The operation consisted of removing the two engines during a 2-hr period a day after the contaminating event. The following day the engines were inspected and repaired, the aircraft was inspected and minor adjustments made during two 3-hr working periods. On the third day the last engine was reinstalled. Engine removal requires the presence of four men. Two work in the vicinity of the front main bearing section (see Fig. 7). These men work from the floor of the hangar. The third man is on the top of the aircraft wing working over the center main bearing, combustion section and rear main bearing. The fourth person is only necessary during the actual removal of the engine at which time he operates the crane.

The average γ-field around the aircraft dropped from about 20 mr/hr the first day to about 6 mr/hr the last day. About 1 ft from the surface of the exposed engine the γ-field was 100 mr/hr at the outset, dropping to about 35 mr/hr at the end. If five of the men divided their time equally between the aircraft and the engine and if they put in 2 hr on the first day and six on the second, their predicted doses would be about 360 mr. This is somewhat higher than 55–200 mr actually measured (Table 2), reflecting self-shielding, geometry factors and "off" time. Likewise the other three men who worked five additional hours on the third day would have been expected to have received 460 mr. Their measured doses ranged from 180–330 mr.

These men did not work on other contaminated aircraft for a week or two before and after

Table 2. Personnel doses to aircraft mechanics on engine removal and aircraft inspection

Name	Dosimeter total (mr)	Total working time (hr)	Position
Cannedy	330	13	Crew chief
Russell	310	13	Engine specialist
Knight	180	13	Engine specialist
Shandy	160	8	Engine specialist
Allen	100	8	Hydraulic specialist
Loche	150	8	A/C mechanic
Sorrell	55	8	A/C mechanic
Dickinson	200	8	A/C mechanic

this event, so that their weekly permissible dose of 300 mr was not exceeded. Thus, with an aircraft that had been engaged in cloud penetration and which had been disassembled in almost the shortest possible time after the contaminating event, the workman who had to execute a complete checkup and engine removal received no more than 330 mr and on the average received less than 200 mr for the entire operation which extended over three days. Similar workmen who performed maintenance on aircraft during Operation Plumbbob and who were responsible for all necessary servicing during the test period received less than the maximum permissible dose for the 2–3 month period. This certainly represents the worst case, as these men were fully occupied with service functions during the period. Had these aircraft been sent to a maintenance depot for service and set aside for a few months, the personnel doses from external radiation for similar work would have been a factor of 10–100 less. In such a case the aircraft could have been serviced without radiation control.

It is of interest to calculate the probable situation from contamination by flights into the normal stratosphere. The preliminary results obtained in this study indicate that aircraft flying in the stratosphere will pick up on the leading edges about 4 dis/min of Sr^{90} per ft^2 per hr of flight. If it is assumed that the average age of the stratospheric debris is one year and that the aircraft flies in the stratosphere for 100 hr before major service, the γ-field associated with this contamination at the leading edge near the fuselage would be less than 10^{-4} mr/hr. Thus, even for protracted flying above the tropopause an aircraft would not be expected to be contaminated to such an extent that the external radiation hazard would be significant (i.e. 0.5 mr/hr, the maximum uncontrolled field for a 40-hr work week).

Some consideration of the external β-hazard is necessary. At 1 ft from the aircraft the β-contribution to the total radiation is small. At 1 in. however, as shown by the data of Table 1, the β-field may be ten times the γ-field. This does not mean, of course, that the surface dose to the total skin of a workman will be ten times the whole body γ-dose. (In such a case the β-hazard to the total skin would be controlling.) As a result of the shielding from their clothing and geometry factors the average β-dose to the total skin should not exceed the γ-dose to the whole body. The other situation to consider is the dose to the bare hands, assuming the workman has his hands constantly on the contaminated surface of the aircraft during the work period. The highest β–γ ratio in Table 1 is about 10. Since the permissible dose to the skin of the hands and the forearms is eight times the whole body γ-dose and since the hands certainly will not be on the aircraft surface during most of the 8-hr work period, it would appear that, in general, the γ-dose to the whole body will be the dominant hazard from external radiation. In certain special cases, the β-dose to the skin of the hands might be controlling. If such cases can be defined the simple requirement of gloves would reduce this hazard below that due to the external γ-dose.

Inhalation

The fission debris may be picked off the aircraft surface by wind currents and enter the surrounding atmosphere. The particle size is small, probably less than 1 μ, so that relatively little of the resuspended contaminant will be removed in the upper respiratory tract. Since the precise particle size distribution under these conditions is not known, it must be assumed that optimum retention occurs. BAIR *et al.*[12] have shown that $Ru^{106}O_2$ on particles ranging from 0.01 to 4.0 μ gave 24 per cent exhaled, 62 per cent deposited in the upper respiratory tract and swallowed, and 23 per cent deposited in the lungs for a set of mice. Since the critical isotopes, iodine and strontium, are in relatively soluble form it is evident that most of these isotopes that are inhaled will be taken into the body. Therefore, it is assumed in the following discussion that the fraction to the critical organ estimated by the NCRP is valid for the debris in air from contaminated aircraft.

The hazard due to inhalation of mixed fission products derived from contaminated surfaces of aircraft can be compared with that from the external γ-radiation. The data from contaminated aircraft both in the hangar and on the

ramp are summarized in Table 3. For each set of experiments the position of the air sampler relative to the aircraft, the time after the shot (to give the age of the debris), the γ-field at that point and the measured air level in total dis/min per m^3 are given. The per cent of the continuous maximum permissible 40-hr work week γ-exposure is from the ratio of the observed value to that calculated from Fig. 3. The per cent of the maximum permissible continuous air exposure is obtained from the ratio of the measured air concentration to the maximum permissible air level for continuous exposure on the 40-hr work week basis for the given age of mixed fission products in Fig. 4. For both of these cases the values are given for a controlled area.

For the first 10 hr after the shot the ratio of the per cent of the maximum permissible γ-dose (MPγ) for a 40-hr work week in a controlled area to the per cent of the maximum permissible concentration of the fission products in air (MPA) ranges from about 2 to 30 based upon experiments conducted at three of the Redwing tests. The only lower ratios during the first day were backgrounds taken at sufficient distance from the aircraft so that the γ-field was very low. The debris, however, could be and was blown in this direction so that the MPγ/MPA must decrease. Likewise in the cockpit the aircraft body contributes appreciable γ-shielding without decreasing the air activity appreciably, since the contaminated air does not originate in the immediate environment. On the second day, whether on the ramp (no. 1) or in the hangar (nos. 2 and 3) the ratio of the %MPγ to %MPA increased. This is the situation except in the closed cockpit and at positions some distance from the aircraft such as the ground under the nose (no. 1) where the γ-level was reduced. On the third to fifth day (nos. 3 and 4) the ratio ranges from 50 to 9000 with the highest value being found in the tailpipe. After several weeks the ratio is still in the range 10 to 1000 with the cockpit remaining low. It is noteworthy that after several weeks even the γ-level is below that for which continuous work is permitted in a controlled area in all cases. The air contamination at the same time is less than that for an uncontrolled area by at least a factor of 10.

These results may be extrapolated to the oldest debris expected, i.e. 1 year, assuming that debris that has been on the aircraft for a period of months is just as available to the air as fresh debris. This is certainly conservative, since the availability to air should decrease with time as the more loosely bound debris is blown off. The ratio of β- to γ-radiation does not change greatly after several weeks. The known fission product decay from 20 days to 1 year would decrease the γ-field given in Table 3 (for the several-week-old debris) by a factor of 15. The potential airborne contaminant in terms of total β-activity would decrease by about the same amount. Likewise the MPγ and MPA would decrease together so that the ratio of %MPγ/%MPA would remain much as it was after three weeks.

A further check is obtained from the data of Plumbbob No. 1 (Table 3). The actual air level around the aircraft in total β-count at 3 to 5 days averaged around 300 dis/min per m^3. The Sr^{90} should therefore have been on the order of 0.03 dis/min per m^3 which is the average value actually obtained from the analysis of the air filters used in the Plumbbob experiments.

The important conclusion may therefore be reached that whether in the hangar or on the ramp, whether contaminated with young or old debris, the potential external radiation hazard will be greater than the inhalation hazard. Practically, therefore, in the case of aircraft contaminated in flight by mixed fission products the inhalation hazard may be disregarded.

Ingestion

In order to compare the ingestion hazard with that from inhalation or external γ-radiation, the efficiency of transfer of the contaminant from the aircraft surface to the mouth must be known. This was investigated by means of a series of experiments in which a person purposely contaminated his hands to the maximum extent by rubbing them vigorously over the aircraft surface. With considerable effort the level of contamination on the hands could be made to approach the surface contamination on the area per unit area. This was followed by handling cigarettes and bread, the two most likely modes

Table 3. Comparison of air concentration and γ-fields near contaminated aircraft

Location	Time after shot	γ-Field (mr/hr)	Air level total β-count (dis/min per m^3)	% continuous 40 hr week γ-exposure	% continuous air exposure controlled	%MPγ / %MPA
Eniwetok Background	3 days	<0.1	10^2	—	0.002	—
Redwing No. 1						
Parking area 100 ft from A/C downwind	2 hr	0.3	500	0.02	0.05	0.4
3 ft from leading edge	2.5 hr	150	6000	9	0.6	15
and fuselage	3 hr	125	2800	10	0.3	33
1 ft inside engine cowling	3.5 hr	250	38,000	25	4.0	6
Parking area 100 ft from A/C	5 hr	—	5100	—	0.6	—
6 in. to leading edge of wing outboard	7 hr	16	1300	4	0.2	20
Parking area	7 hr	—	3800	—	0.5	—
2 ft above leading edge	7.5 hr	10	2700	2.5	0.4	6
Parking area	9 hr	—	1400	—	0.2	—
Closed cockpit	28 hr	2	7600 (?)	2	1.3	1.7
Under nose	32 hr	3	1100	3	0.2	15
Parking area	33 hr	—	1500	—	0.2	—
Below wing	34 hr	2	280	2	0.05	40
Closed cockpit	36 hr	2	640	2	0.11	19
Parking area	38 hr	—	460	—	0.08	—
Redwing No. 2						
2 ft from nacelle	2 hr	500	43,000	29	4.3	7
Inside nacelle	2.5 hr	4200	175,000	250	17.5	14
Open cockpit	4 hr	100	18,000	10	2.0	5
Off ramp	4 hr	4	23,000	0.4	2.5	0.2
Deep in airscoop	5 hr	4000	351,000	710	44	16
Closed cockpit	6 hr	80	59,000	18	7	3
Off ramp	7 hr	3	240	0.75	0.03	25
Closed cockpit (hangar-engine study)	9 hr	22	17,500	7	2.1	3
Main-bearing cooling air duct	36 hr	>170*	57,000		9	19
Exhaust duct and rotary duct	37 hr	310	15,900	340	2.6	130
Engine well in wing	38 hr	43	4800	48	0.8	60
Redwing No. 3						
In nacelle	2.5 hr	2500	935,000	125	94	1.4
Ramp background	2.5 hr	25	7300	1.2	0.7	1.8
Inside cockpit	3 hr	320	45,000	17	4.5	4
2.5 ft from left nacelle	3 hr	400	39,000	21	4.0	5
Ramp background	4 hr	17	1400	1.7	0.16	11
Deep in airscoop	5 hr	800	110,000	193	13	15
Cockpit	6 hr	120	13,000	30	1.6	19
Ramp background	7 hr	9	1800	2.3	0.2	12
In nacelle	9 hr	700	76,000	230	9	26
2 ft from wing root	11 hr	170	5800	65	0.7	93
Ramp background (hangar-engine)	11 hr	6	600	2.3	0.08	30
Base of engine	33 hr	100	4000	110	0.7	160

* Air sample taken inside the compressor where the gamma field was undoubtedly higher but gamma survey instrument could not be placed in this position. Reading shown is from outside.

Table 3 (*contd.*)

Location	Time after shot	γ-Field (mr/hr)	Air level total β-count (dis/min per m^3)	% continuous 40 hr week γ-exposure	% continuous air exposure controlled	$\frac{\%MP\gamma}{\%MPA}$
Background	33 hr	17	2400	19	0.4	48
Horizontal stabilizer	34 hr	20	600	22	0.1	220
Engine intake	34 hr	70	1300	77	0.2	390
Exhaust cone	35 hr	130	9700	160	1.6	100
Background	35 hr	12	1500	13	0.25	52
Compressor section	36 hr	80	22,000	89	3.7	24
Background	50 hr	6	700	9	0.1	90
Air bleed	50 hr	40	1200	57	0.2	285
Air cond. ducts	51 hr	240	12,000	340	2.0	170
Background	51 hr	9	1200	17	0.2	85
Air cond. well	52 hr	90	500	130	0.1	1300
Background	54 hr	5	500	7	0.1	70
Engine during reinstallation	56 hr	60	800	86	0.1	860
Plumbbob No. 1						
Cockpit	3 days	2	336	4	0.07	57
Behind cockpit	3 days	8	216	16	0.06	267
Top of wing	3 days	20	384	40	0.08	500
Tail radar cone	5 days	3	374	9	0.07	130
Tail pipe	5 days	~200	370	625	0.07	8900
Outside mid-engine	5 days	18	350	56	0.09	620
Engine intake	5 days	25	2100	78	0.5	156
Exhaust cone	5 days	12	2820	37	0.7	53
Turbine blades	5 days	11	1600	34	0.4	85
Plumbbob No. 2						
Front compressor blades	5 days	25	703	78	0.2	390
Outside mid-engine	5 days	9	1180	28	0.3	93
Combustion intake	5 days	20	2817	63	0.7	90
Plumbbob No. 3						
Cockpit	3 weeks	0.3	330	2.1	0.2	11
		0.2	330	1.4	0.2	7
Rear cockpit	3 weeks	0.6	648	4.3	0.4	10
		0.5	90	3.6	0.06	60
Top wing	3 weeks	0.9	1030	6.4	0.6	10
Tail pipe	3 weeks	2	762	14	0.5	28
		1	903	7	0.5	14
Outside mid-engine	3 weeks	10	340	70	0.2	350
		6	653	43	0.4	110
		7	126	50	0.1	500
		5	514	36	0.3	120
Intake blades	3 weeks	4	456	29	0.3	97
		5	629	36	0.4	90
		12	126	86	0.1	860
		7	252	50	0.2	250
Exhaust	3 weeks	10	130	70	0.1	700
		12	149	86	0.1	860
Tail cone	3 weeks	2	680	14	0.4	35
Turbine blades	3 weeks	3	456	21	0.3	70
Tail	3 weeks	2	49	14	0.03	470

Table 4. Comparison of Sr^{90} contamination of gloves with bare hands for similar work on the aircraft Zuni

Length of service operation	Date	Surface	γ-Field (mr/hr)	Sr^{90} dis/min per pair of gloves or hands
4 hr	5/30/56	gloves	8	17,000
$3\frac{1}{2}$ hr	5/30/56	gloves	7	14,100
$2\frac{1}{2}$ hr	5/30/56	gloves	5	2800
4 hr	5/30/56	gloves	5	3000
4 hr	5/30/56	gloves	5	4300
4 hr	5/30/56	gloves	5	6100
Wipe over aircraft	5/29/56	hands	20	420
Wipe over aircraft	5/29/56	hands	20	400

of transfer to the body. The cigarette experiment consisted of rolling each cigarette of a full pack individually between the palms of the hands. The bread was handled as a sandwich. In other cases it was completely crumbled in the process of dry "washing" the hands. The experiments conducted during Redwing indicated that in the course of normal work the hands would only reach about 40 per cent of the contamination level on the aircraft surface. Similar experiments during Operation Teapot resulted in hand contamination that was only 30 per cent of the level on the aircraft surface even after vigorous rubbing. To be conservative it will be assumed for purposes of calculation that on the average the workman will contaminate his hands to a level of about half that of the aircraft surface on which he is working.

In several cases gloves were worn. In general, much more contamination was picked up on the gloves than on the bare hands. In Table 4 the quantity of Sr^{90} obtained from analysis of the gloves is compared with the amount on the bare hands for a similar operation.

After removing the gloves the men who had received 17,000, 14,000 and 2800 dis/min Sr^{90} on the gloves washed their hands with bread yielding 66, 19 and 8 dis/min Sr^{90}, respectively. No special precautions were used to remove the gloves so that this small contamination came largely from the handling of the gloves, since in other cases where great care was taken in removing the gloves no contaminant was observed on the hands. Aside from the problem of abrasion it would appear that the use of soap and water at the end of the shift would be more practical than the use of gloves. With special precautions for removal the gloves would prevent an appreciable hand contamination while at the same time they would protect the skin from β-radiation. Thus, if the ingestion hazard turned out to be significant, the simple expedient of wearing gloves would reduce it to negligible proportions. This could be easily accomplished in controlled areas. Since it is desired to evaluate the hazard in uncontrolled areas, however, the efficiency of transfer by bare hands must be defined.

Once the contaminant is on the hands, it may pass into the body on food which is handled or on cigarettes which are smoked. In general it was found from a dozen experiments on different tests and different individuals that rolling a pack of cigarettes in the hands and handling them extensively transferred about 10 per cent of the surface contaminant. Normal handling transferred only a few per cent of the contaminant from the hands to the cigarette. Experiments at Oak Ridge[13] showed that normal handling of a contaminant consisting of dry uranium compounds adhering to the palms or surfaces of the hands, transferred only 1 per cent to the cigarette. They found further that only 0.2 per cent of this 1 per cent may appear in the smoke. The most important transfer would then presumably occur from the area of the cigarettes actually touched by the lips. This would not exceed one tenth of the surface area

of the cigarette. Thus, if a pack of cigarettes per day were smoked in the normal way during work with contaminated hands, the quantity ingested would not exceed 0.2 per cent of that on the hands if a total transfer of 2 per cent is assumed and the lips are 100 per cent efficient in removing the contaminant from the area they touch. Both of these assumptions are conservative.

Handling several pieces of bread as a sandwich with slightly moist hands transferred about 20 per cent whereas "washing" the hands in bread by crumbling removed 50–80 per cent. In the worst case a workman might eat lunch and dinner without washing his hands. This would result in removing about 20 per cent of the hand contamination twice, or a total of 40 per cent. The bread will presumably be eaten so that all of the 40 per cent of the hand contamination will be ingested. The cigarettes would presumably be smoked at various "breaks" throughout the day. Normally, practically all surface contamination on the body will be washed or rubbed off in 24 hr so that only the day of the contamination needs to be considered.

In order to compare the ingestion hazard with that from external radiation all pertinent data are compiled in Table 5. The average γ-field is obtained from the surveys in the area that the men were working. The $\%MP\gamma$ is obtained from Fig. 3 for the controlled 40-hr

Table 5. Comparison of ingestion hazard and γ-field

Event	Time after shot	Average γ-field (mr/hr)	$\%MP\gamma$	Sr^{90} ingest. experimental (dis/min per day)	Total β-count ingested (dis/min)	%MPI	$\frac{\%MP\gamma}{\%MPI}$
Redwing No. 1							
(nose–wing area)	28 hr	9	8	9	2.2×10^5	5	1.6
Redwing No. 2							
(nose area)	34 hr	20	22	84	2.1×10^6	50	0.4
				8	2.0×10^5	5	4
				19	4.8×10^5	13	1.7
				67	1.7×10^6	45	0.5
Redwing No. 3							
(nose area)	26 hr	26	24	5	1.4×10^5	3	8
				18	5.6×10^5	15	1.6
(engine)	52 hr	50	78	18	2.1×10^5	5	16
				100	1.2×10^6	30	2.6
				11	1.3×10^5	3	26
(engine)	56 hr	50	78	362	4.3×10^6	120	0.7
				103	1.2×10^6	30	2.6
				850	1.0×10^7	300	0.3
Plumbbob No. 1							
(engine area)	3 days	10	23	4	4×10^4	1.3	18
		25	57	43	4.3×10^5	14	4
		25	57	34	3.4×10^5	11	5
		25	57	21	2.1×10^5	7	8
Plumbbob No. 2							
(engine area)	20 days	8	69	19	1.9×10^4	0.6	115
				54	5.4×10^4	1.8	38
				103	1.0×10^5	3	23
Plumbbob No. 3							
(engine area)	20 days	3	21	16	1.6×10^4	0.5	42
				45	4.5×10^4	1.5	14
				16	1.0×10^5	3	7

work week. The Sr^{90} that might be ingested per day, assuming that the hands are not washed, is obtained from the actual analytical results on the bread which had been handled in the normal manner. (The "washing" of hands with bread was not included here, as it represents an extreme and unlikely case.) From the known concentration of Sr^{90} in mixed fission products at the time of the experiment, the total β-radiation ingested can be calculated. Then, from the curve for a controlled area (Fig. 5) the %MPI (maximum permissible daily ingestion) can be calculated. The ratio of the %MPγ to the %MPI indicates the relative importance of the two sources of radiation hazard. It is seen in general the external radiation hazard is more important than the ingestion hazard even with fresh debris. After 3 days, the γ-hazard becomes ten to a hundred times more significant than the ingestion hazard.

The low values in the vicinity of the engine at 56 hr on the Redwing No. 3 experiment are probably due to the fact that the γ-field listed was only the general level in the vicinity of the engine. If closer monitoring of the actual work position had been done, the γ-fields and hence the %MPγ/%MPI would have been higher. Again, since both the MPI and the MPγ decrease with time together, the γ-effect will remain the dominant one forever. This analysis, of course, takes into account the I^{131} problem by referring everything to the *maximum permissible total* β-count at any given time.

In a number of these experiments both bread and cigarettes were handled, but, as discussed above, the potential ingestion due to smoking a pack during the work day would be less than that taken in from eating by a factor of 20 or more. It must be emphasized that the ingestion experiment on which this conclusion is based is very conservative since it is normal for men to wash their hands before eating and with this very simple remedy the potential ingestion is essentially reduced to zero. Certainly in a controlled area the men would be required to wash their hands routinely.

It is also important to note that even by 20 days after detonation on a typical case of cloud penetration the actual ingestion level is 100 times less than the MPI. The γ-field is one-quarter to one-half of the MPγ for continuous work in a controlled area. By half of a year this would probably fall below the radiation field permitted for release to an uncontrolled area that had a 40-hr per week occupancy factor. Since these aircraft have been exposed under the worst conditions, it may be concluded, that in virtually all cases the γ-field in the vicinity of a contaminated aircraft 1 year after the contaminating event would be far below the MPγ. Likewise, the daily ingestion to workmen would be far below the MPI.

In view of these results it is not surprising that even under the unsanitary and dirty conditions of life to which the Rongelap natives are accustomed, the external radiation dose was the controlling factor.[14]

In a cloud penetration flight the pilot receives far more radiation due to the external field than from ingestion or inhalation. This was shown experimentally by the Upshot–Knothole study in 1953, Project 4.1. It was confirmed in part here by the background levels of Sr^{90} in the urine and feces of pilots who had received several roentgens during cloud penetration.

Finally, it is of interest to see if any of the maintenance men actually lived during one of the operations without washing hands before meals. For this, urine samples were taken at the end of the test series and analyzed for Sr^{90}. Assume that the men who worked throughout Plumbbob spent 3 days each on ten aircraft that had been contaminated by various shots. Further, assume from the bread handling data that during each of these 30 days they ingested 100 dis/min of Sr^{90}. This would lead to an average bone level[15] of 0.3 $\mu\mu$c Sr^{90}/g Ca. Since the average adult skeleton in the U.S. in 1957 had about 0.25 $\mu\mu$c Sr^{90}/g Ca from fallout Sr^{90} in the natural diet and since the standard deviation around this mean is at least $\pm$40 per cent, it would be difficult to establish clearly this amount of contamination from Plumbbob operations by this method. In the average person the urine level is actually higher than the bone level as it is held up by the daily intake of Sr^{90} in the normal diet. The urinalyses in Table 6 are in accord with this evaluation showing that in all cases the Sr^{90} level can be accounted for by the normal diet. Over long

Table 6. Sr^{90} *in urine, aircraft maintenance personnel, Operation Plumbbob*

Person	Job	Previous radiation exposure (mr)	Plumbbob exposure (mr)	($\mu\mu$c Sr^{90}/g Ca)		
				Arrive at Plumbbob	Interim	End of Plumbbob
1	Dock crew	—	1500	6.5	<1	<4.5
2	Dock crew	6200	1800	2.1	<0.8	<1
3	—	—	—	3.6	—	—
4	Crew chief	4500	1500	<1.4	<5	—
5	Dock crew	—	1050	<0.6	—	5.1
6	—	—	—	6.3	—	—
7	Crew chief	5700	3100	<0.5	—	4.4
8	—	—	—	<1.2	—	—
9	—	—	—	2.5	—	—
10	Crew chief	5170	850	<0.4	9	—
11	—	—	—	<10	—	—
12	Dock crew	—	1290	2.9	—	<3
13	Electrician	—	530	1.9	<1.5	~5
14	Dock chief	4300	1200	6.2	—	2.9
15	—	—	—	1.0	—	—
16	—	—	—	7.1	—	—
17	—	—	5000	—	—	2.1
18	Flight chief	4480	900	12	7.2	4.3
19	—	—	—	2.3	—	—
20	Crew chief	—	2100	6.8	—	—
21	Dock crew	—	1220	<3.0	—	<1.0
22	Dock crew	—	—	<1.0	7.7	1.5
23	—	—	—	<1.4	—	—
		Average (maximum)		3.7	4.6	3.2
		range normal U.S. civilian		—	—	1–10

periods of repeated exposure the urinalysis for Sr^{90} will give a measure of the skeletal level if it is sufficiently above the dietary level.

Monitoring techniques

Since it appears certain that the radiological hazard from servicing aircraft contaminated with mixed fission products is dominated by the external radiation, the monitoring process is greatly simplified. A properly calibrated γ-survey meter is all that is required to define whether or not an aircraft may be released to uncontrolled areas regardless of the age of the debris. This level is 0.25 mrem/hr for a 40-hr work week. Since it has been shown that the actual dose received by workmen is only about half of that indicated by the survey meter, the acceptable level for release should be taken as 0.5 mr/hr on a survey meter. Since the standard survey meters, such as the Air Force AN/PDR-27C, read down to 0.01 mr/hr no special equipment is required. In controlled areas the γ-survey will be supplemented by film badge and pocket dosimeters so that the local health physicist may readily determine the proper number of work hours per week for any given situation.

Swipe sampling has been proposed as a method of estimating: (a) surface contamination and (b) the associated air concentration due to resuspension. Actually it is not preferred for either measurement in the aircraft contamination problem.

Various methods of swiping the surface contamination of the aircraft were conducted both at Redwing and Plumbbob using a variety of techniques. These included a rotary constant-pressure swiper, a rectangular constant-pressure swiper, the conventional manual technique, and stripping the surface with gummed paper and masking tape. In addition to varying the mechanical techniques, the following compounds were applied to the dry filter paper used in the manual or mechanical swipers; water, oil, Gunk, gasoline and HCl. Virtually all of these

Table 7. Comparison of swipe and air sampling

Position	Air activity (dis/min per m^3)	Swipe (1000 dis/min per 100 cm^2)	Swipe/air
A. *Plumbbob No.* 1 (*3 and 5 days after shot*)			
Cockpit	336	12.4	0.04
Cockpit	216	620	2.9
Top wing	384	915	2.4
Tail radar cone	374	1000	2.7
Tail pipe (1)	382	6250	16
Tail pipe (2)	367	5000	14
Mid-section (1)	328	115	0.35
Mid-section (2)	370	15	0.04
Engine intake (1)	1900	840	0.4
Engine intake (2)	2280	680	0.3
Engine cone (2)	2820	780	0.3
Engine turbine (1)	1600	1130	0.7
B. *Plumbbob No.* 2 (*Several weeks after test*)			
Back of rudder	6060	205	0.03
Tail pipe (1)	762	1000	1.3
Tail pipe (2)	903	1144	1.3
Cockpit (1)	330	5.1	0.02
Cockpit (2)	648	87.2	0.13
Top wing	1030	42.0	0.04
Mid-section (2)	653	23.0	0.04
Mid-section (1)	340	14.0	0.04
Engine intake (2)	456	500	1.1
Engine intake (1)	629	140	0.2
Engine tail cone (2)	680	1200	0.2
Engine turbine (1)	456	900	2.0
C. *Plumbbob No.* 3 (*several weeks after tests*)			
Cockpit (1)	288	2.0	0.007
Cockpit (2)	90	500	5.5
Mid-section (1)	514	568	1.1
Mid-section (2)	126	5.5	0.04
Engine intake (1)	252	312	1.2
Engine intake (2)	126	167	1.3
Engine exhaust (1)	130	>5000	>38
Engine exhaust (2)	49	>5000	>102
Tail	49	284	5.8
Front compressor	703	5540	7.9
Mid-engine	1180	900	0.8
Combustion chamber	2817	800	0.3

techniques, with the exception of the manual HCl wash with vigorous rubbing, removed a very small and highly variable fraction of the contaminant. In general, the removal was on the order of tenths of per cent with a rotary constant-pressure device. The variations from one swipe to the next even in this case were large (factors of 2 and 3 being common). The standard manual swipe is at least as variable but does remove a larger fraction of the activity (1–10 per cent).

A series of experiments involving eight aircraft, each contaminated by a different event, was conducted in which the concentration of the fission products in air was determined immediately adjacent

to a surface which was swiped in the normal way, i.e. manual rubbing of 150 cm^2 with a dry filter disk. Table 7 summarizes these data and shows that the ratio of swipe activity (in 1000 dis/min per 100 cm^2) to the air activity (in dis/min per m^3) varies greatly from 0.007 to greater than 102. It is evident that there is no systematic relation between these quantities. Aside from the great variability in the fraction of contaminant removed in the swiping, there is the fundamental reason that the concentration of the contaminant in the air is simply not related to the surface activity of a small area but reflects the emission of the debris from all parts of the contaminated aircraft.

The swipe measurement is not particularly useful even for estimating the potential ingestion hazard from contaminated aircraft. The swipe gives no more reliable an estimate of the ingestion hazard than does a mere γ-survey. The best method is a close (1 in.) β-survey. Table 8 gives the comparison of the β-activity on aircraft surface by direct β-measurement and by standard swipe sampling. In these instances it is seen that the swipe technique removes 1–2 per cent of the contaminant on the average but that it may vary by a factor of 10. The more highly contaminated surfaces show a lower efficiency of removal averaging about 0.4 per cent with a smaller standard deviation than is the case with the less contaminated surfaces. This may reflect the increased accuracy of measurement at higher levels of activity. There is no evidence that the variation in the apparent fraction removed by swiping is due to real differences in the adherence of the contaminant at various locations on the aircraft surface. It is concluded that the surface β-activity of a small area on an aircraft contaminated by an atomic cloud can best be estimated by a direct β-measurement rather than a swipe technique.

SUMMARY AND CONCLUSIONS

(1) The radiation hazard to maintenance personnel who must work on aircraft contaminated with mixed fission products can be defined adequately by the γ-field in which they will work. The contribution to their total radiation dose from inhalation of resuspended debris or ingestion from hand contamination is generally small. After several days the ratio of the external γ-dose to that obtained from inhalation or ingestion is 10 to 1000. For year-old debris the levels for aircraft purposely flown through fresh clouds will be far below the MPγ or MPI for release to uncontrolled factory areas.

(2) The γ-field around aircraft which have purposely penetrated an atomic cloud is such that maintenance men can work on it for about 8 hr in the second to fourth day after detonation without exceeding their weekly dose. After 3 weeks the γ-level except at the engine is low enough to permit release to uncontrolled areas. If no working position exceeds 0.5 mr/hr the aircraft can be released to an uncontrolled area.

(3) The acceptable level for a workman from a radiation survey of an aircraft should be taken as twice the permissible dose rate in rems since experiment has shown that the actual dose is about half that of the air flux as a result of self-shielding, geometry and work pattern.

(4) The γ-level for aircraft which spend most of their flight time in the stratosphere will probably never exceed the uncontrolled level unless a fresh cloud is encountered or the stratospheric burden of fission products is greatly increased.

(5) The surface contamination and the γ-field on or near a contaminated aircraft can be estimated from a survey of the leading edge of the wing. The γ-field around the aircraft varies only by about a factor of 10 being highest near the engines and the body.

(6) The radiological hazard from contaminated aircraft is essentially independent of the type of nuclear explosion.

(7) The maximum permissible air concentration (MPA) of total β-activity from mixed fission products in controlled areas for a 40-hr work week varies from 10^6 dis/min per m^3 at 2 hr to 6.5×10^5 dis/min per m^3 at 1 day to 1×10^4 dis/min per m^3 at 1 year.

(8) The maximum permissible ingestion level (MPI) in terms of total β-activity (dis/min per day) for mixed fission products varies from 4×10^6 dis/min per day at 1 day, 2.6×10^6 dis/min per day at 30 days and 2×10^5 dis/min per day at 1 year for controlled areas.

(9) Analyses of urine and feces of men who obtained up to 10,000 mr during an operation or those who worked for from 2 to 3 months on contaminated aircraft showed no detectable Sr^{90} ingestion above background levels. This result was expected from calculations of the potential ingestion and inhalation during these operations.

(10) Swipe sampling is highly variable. The

Table 8. Comparison of β-activity on aircraft surface by direct measurement and swipe sampling
(The β-activity at 1 in. is taken as the actual surface value; values given in dis/min per 10 cm^3 × 10^{-6})

Position	β-measurement	Swipe sample	% swipe removal
Plumbbob No. 1—3 to 5 days after test			
Cockpit	3	0.01	0.3
Fuselage	27	0.62	2.3
Wing	50	0.92	1.9
Tail cone	73	1.0	1.3
Engine intake	145	0.84	0.6
	260	0.68	0.3
Turbine blades	215	1.13	0.5
	270	0.78	0.3
Exhaust pipe	1500	5.0	0.3
	4000	6.3	0.2
			—
		Avg.	0.8
Plumbbob No. 2—week after test			
Cockpit	0.3	0.005	1.7
Fuselage	0.9	0.087	10
Wing	2.0	0.042	2.0
Outside engine—			
mid-section	2.0	0.014	1.4
	3.0	0.023	0.8
Back of rudder	3.8	0.21	5.4
Intake blades	62	0.50	0.8
	81	0.14	0.2
Tail cone	180	1.2	0.7
Turbine blades	211	0.900	0.4
Exhaust pipe	195	1.14	0.5
	590	1.00	0.2
			—
		Avg.	2.0
Plumbbob No. 3—Several weeks after test			
Cockpit	0.1	0.002	2
Fuselage	3.5	0.500	14
Outside engine—			
mid-section	1.0	0.005	0.5
Tail	5.0	0.568	10
Engine intake	7	0.284	4
	18	0.312	1.7
	60	0.167	0.3
Exhaust	660	>5.0	>0.8
	640	>5.0	>0.8
			——
Plumbbob No. 4		Avg.	3.8
Outside engine—			
mid-section	10	0.9	9
Intake	260	0.800	0.4
Compressor blades	1600	5.5	0.3
			—
		Avg.	3.2

Table 8 (*contd.*)

Position	β-measurement	Swipe sample	% swipe removal
Redwing No. 3			
Side fuselage (2)	17	0.15	0.9
Under wing (2)	94	0.53	0.6
Tail pipe (2)	402	1.4	0.35
Bomb-bay door (2)	33	0.87	2.3
Wing top (2)	22	0.5	2.2
Tail pipe (2)	340	1.1	0.3
Fuselage by inverter	7	0.036	0.5
Access door	7	0.085	1.1
Bearing section exterior	30	0.61	2.0
Combustion section exterior	0	0.014	—
Exhaust (2)	320	1.4	0.4
Combustion section exterior	0	0.047	—
Bearing	14	0.60	4.3
			—
		Avg.	1.4

standard dry swipe averages 2 per cent efficiency but the range from point to point may be a factor of 10. No difference is obtained if water, gasoline, Gunk or dry filter paper is used. HCl will remove most of the activity apparently by dissolving a thin film of the aluminum and aluminum oxide into which the particles are embedded.

(11) There is no correlation between air and swipe sampling since the concentration of debris in the air is not determined by the activity level on the surface immediately adjacent to the sampling site. There is a general correlation between the swipe sample and the removable activity but β-surveying is more reliable and much simpler.

(12) Workmen will normally acquire a hand contamination level less than half of that on the surface on which they are working. Approximately 20 per cent of this will be ingested if they eat a sandwich-type meal without washing their hands. If they wash their hands, or wear gloves and take precautions on removing them, the potential ingestion hazard is entirely negligible. The quantity that could be ingested by smoking a pack of cigarettes per day during working hours without washing the hands would be at least a factor of 20 less than that from eating.

(13) Measurements of Sr^{90} contamination from aircraft flying in the stratosphere indicate that the long term contamination will not present an important ingestion hazard to maintenance personnel.

(14) α-contamination is not a significant hazard from aircraft contaminated by normal fission explosions. Measurements indicate that it is much less than the Sr^{90} ingestion hazard under the same conditions.

(15) The radiation hazard from aircraft contaminated with mixed fission products does not require procedures which will significantly impair operational efficiency nor should it increase the cost of aircraft maintenance in order to protect the workmen fully.

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At Operation Redwing Captain PAUL CRUMLEY helped in many ways, not the least of which was morale, before his untimely death in the course of duty.

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SORPTION OF CESIUM BY CONASAUGA SHALE

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Abstract—Disposal of intermediate-level wastes at ORNL involves the discharge of the waste into a series of three 1 million-gal pits excavated in a Conasauga shale formation. The activity of the waste is about 1.9×10^6 dis/min per ml, which is too great for direct release to the uncontrolled environment. The solution is essentially basic sodium nitrate with cesium-137 contributing the majority of the activity.

Conasauga shale has been found to be effective for removal of cesium from the waste stream. The partial cesium capacity of the shale was found to be dependent on the nature and concentration of the major stable cations present in the waste. The capacity of the shale for "fixation" of cesium greatly exceeds the capacity of the shale for sorption of cesium from the waste stream; thus, the cesium retained from the waste stream would be very difficult to remove from the formation by percolating ground water. Because fixation sites incorporate charge deficiency with structural parameters, removal of cesium from the shale is increased by increasing the concentration of cations in the leaching solution.

INTRODUCTION

Soils exhibit a marked capacity for retaining cations in a loosely bound or exchangeable state. This ability of the soil to remove and retain cations from aqueous solutions has been the basis for disposal of large volumes of intermediate-level fission product wastes at ORNL for a number of years.[1,2] During the period June 1952 to September 1958, seepage pits constructed in a Conasauga shale formation have been used for the disposal of 11×10^6 gal of such wastes. These waste pits presently accommodate approximately 3×10^6 gal of this type waste per year.

Although performance of the seepage pits under field conditions is influenced by factors of porosity, permeability and hydraulic gradient, one of the more important considerations regarding the management of the waste pits concerns the partial capacity of the Conasauga shale for the sorption and retention of the radioactive constituents in the waste.

Analysis of the major radionuclides and stable chemical constituents of the waste liquid in the pits, Table 1, shows that the waste liquid is essentially basic sodium nitrate. Cesium-137 with a half-life of 30 years and a high fission yield contributes 85 per cent

Table 1. *Analysis of a grab sample of radioactive liquid waste from pit* 3, *collected* 19 *October* 1956[3]

Ionic composition		Radioactive composition	
Ion	Concentration (mg/ml)	Radionuclide	Concentration (μc/ml)
Na^+	14	Cs^{137}	0.66
Al^{3+}	0.29	Ru^{106}	0.094
K^+	0.15	Co^{60}	0.007
Si^{4+}	0.020	Sb^{125}	0.004
NH_4^+	0.002	Sr^{90}	0.007
Fe^{3+}	0.001	Rare earths and yttrium	0.007
Ca^{2+}	0.075		
NO_3^-	21	Zr^{95}	0.001
OH^-	2.9	Nb^{95}	0.013
SO_4^{2-}	2.1		
CO_3^{2-}	1.8		
PO_4^{3-}	0.20		
Cl^-	0.18		

* Operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

of the gross β-activity in the waste. Ruthenium is the second greatest contributor with strontium-90 being present in smaller amounts. Because of the abundance of cesium-137 in the waste, a detailed study of its behavior was undertaken.

EXPERIMENTAL

Equilibrium studies with cesium

A sample of the light-colored, weathered portion of the Conasauga shale was collected, dried and ground to pass a 20-mesh sieve. Mineralogical analysis of the Conasauga shale from the waste pit disposal area by the U.S. Geological Survey revealed that the shale is heterogeneous in nature, though illite is the predominant clay mineral in the light-colored, weathered portion of the shale. The cation-exchange capacity was determined by the ammonium acetate method and found to be 16.5 mEquiv/100 g.[4] Two hundred grams of this shale was allowed to stand for 24 hr in 1 l. of 1 N chloride salt solution of the appropriate complementary cation. The shale suspension was transferred to a Buechner funnel and leached with an additional 2 l. of the 1 N salt solution. The shale was then elutriated with ethanol until the effluent gave a negative test for chloride.

Six grams of the homoionic shale, representing 1 mEquiv of exchange capacity, was weighed into a bottle containing 100 ml of tagged $CsNO_3$ solution and of the desired cation concentration. The concentration range of cesium was varied from 10^{-7} M (10^{-5} mEquiv/100 ml) to 0.02 M (2 mEquiv/100 ml). After allowing 65 hr for completion of the exchange reaction, 1-ml aliquots of the supernatant were removed for counting on a well-type scintillation counter. The percentage cesium sorbed was estimated by comparing the final and initial counting rates, after they had been corrected for background and volume changes.

The results showing the influence of the valence of the complementary cation are plotted in Fig. 1. The data show that in the more concentrated solutions of cesium, sorption of cesium is decreased with increasing valence (Na^+, Ca^{2+}, Al^{3+}) of the competing cation. However, in dilute solutions of cesium, the valence of the complementary cation was found to have little influence on the sorption of cesium. In this case the specific bonding energy for cesium overshadows the increased affinity of the charged surface for counter-ions of higher valence. Small differences in cesium sorption in this concentration range are difficult to observe as the sorption is nearly complete.

The effect of ionic radius of the complementary cation was also investigated. Fig. 2 shows that cations having an ionic radius similar to that of the cesium ion are most effective in competing with

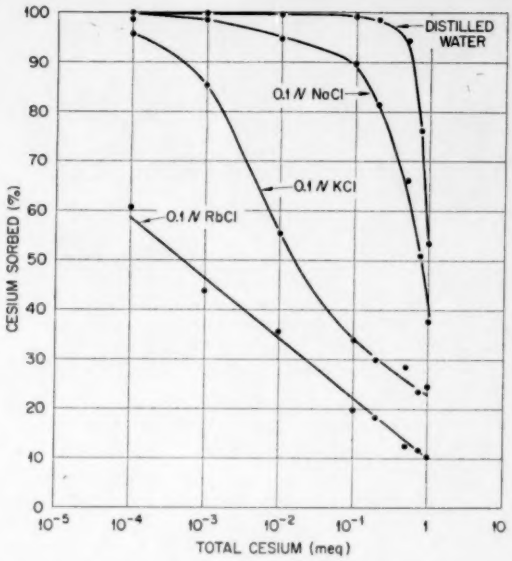

FIG. 1. The effect of the ionic radius of the complementary cation on the retention of cesium by 6 g of Conasauga shale.

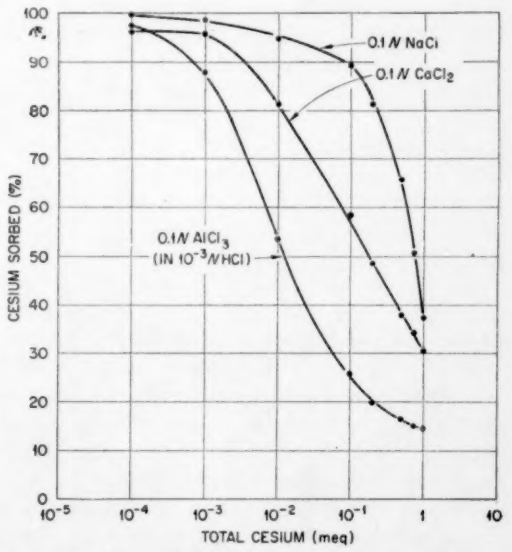

FIG. 2. The effect of the complementary cation valence on the retention of cesium by 6 g of Conasauga shale.

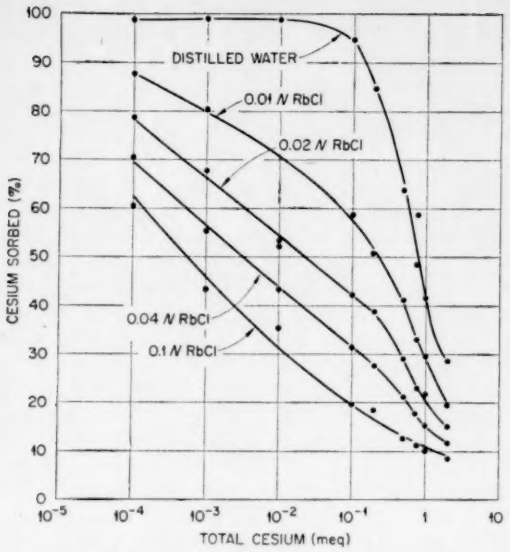

FIG. 3. The effect of rubidium concentration on the retention of cesium by 6 g of Conasauga shale.

cesium for the exchange sites that are structurally optimum for cesium sorption. The non-hydrated radii of the cations employed are $r_{Cs} = 1.65$ Å, $r_{Rb} = 1.49$ Å, $r_K = 1.33$ Å and $r_{Na} = 0.98$ Å.

Since a limited number of exchange sites are available, there is competition among the ions for these sites. The law of mass action predicts a decreased sorption of cesium with increasing concentration of a competitive cation, such as rubidium (Fig. 3). Increasing the concentration of cations that are less effective in competing with cesium for "fixation sites" results in a less marked inhibition of cesium sorption as was shown in Figs. 1 and 2.

As the cesium concentration is decreased a greater proportion of cesium ions sorbed by the shale are held at "fixation sites", and the selectivity for cesium compared to other cations increases. Table 2 is a tabulation of selectivity coefficients computed for the various monovalent cation salt solutions, and it shows the marked dependence of selectivity on cesium concentration. It is seen from these data that the behavior of cesium present as a microconstituent differs considerably from its behavior when present as a macroconstituent.

Table 2. Selectivity coefficients obtained at various levels of cesium concentration in a number of monovalent cation solutions

Total cesium (mEquiv)	Selectivity coefficient = (Cs-sorbed)(M-solution) / (Cs-solution)(M-sorbed)		
	0.1 M NaCl	0.1 M KCl	0.1 M RbCl
2.0	—	—	1.2
1.0	10	3.0	1.3
0.75	17	2.6	1.4
0.50	30	4.7	1.6
0.25	56	4.6	2.4
0.10	93	5.4	2.5
0.01	186	12.4	5.4
0.001	823	60	7.8
0.0001	1950	212	15.3
0.00001	—	—	13.3

Column sorption studies

Since the seepage pits function as a continuous ion-exchange system, it was desirable to investigate the effectiveness of Conasauga shale by column techniques. Increments of 0.23 M $NaNO_3$ solutions containing 0.55×10^{-7} M $CsNO_3$ were passed through 6-g portions of the shale. Results, as seen in Fig. 4, show the total amount of cesium retained by successive shale increments. Forty per cent of the total cesium had remained in solution after 10 l. of waste were passed through the first 6-g increment of the shale. This was reduced to 17 per cent by passing the effluent through a second increment of shale. Passage through the third, fourth and fifth increments of shale reduced the amount of cesium remaining in solution to 6.9 per cent, 1.2 per cent and 0.2 per cent of the total activity, respectively. The partial cesium capacity of the shale for this solution was estimated to be 5.6 μEquiv/100 g from the data of this experiment. This estimate was made by a linear extrapolation of the amount of cesium remaining in solution after passing through the 6-g portion of shale.

When the $NaNO_3$ concentration was increased from 0.23 M to 0.5 M, the partial cesium capacity of the shale dropped to 2.4 μEquiv/100 g. The selectivity coefficient, the Cs/Na ratio on the clay compared to this ratio in the solution at equilibrium, was approximately equal for both solution concentrations, indicating that in a $NaNO_3$ system, the Cs/Na ratio of the solution determines the partial cesium capacity of the shale.

When 0.5 M NH_4NO_3 was employed as the carrier solution, the shale retained only 0.03 μEquiv of cesium per 100 g. This extreme drop in the partial cesium capacity can be attributed to the

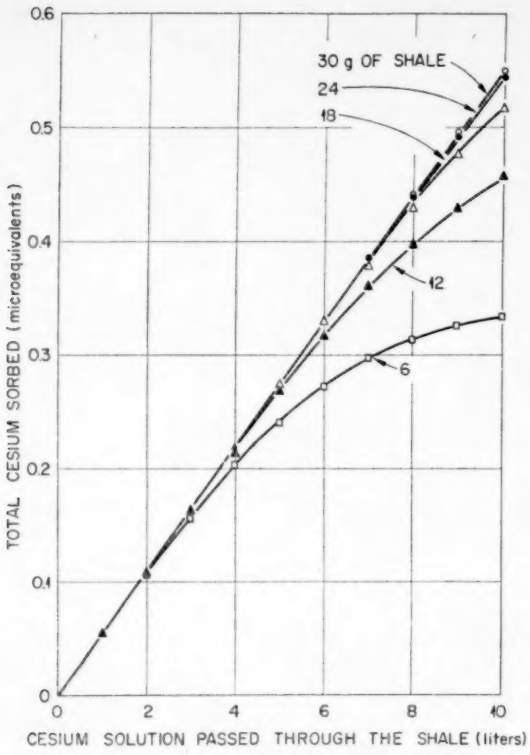

FIG. 4. Sorption of cesium from flowing solutions by Conasauga shale.

ability of the NH_4^+ ion ($r_{NH_4} = 1.59$ Å) to compete effectively with cesium ions for the "fixation sites".

Desorption studies

In addition to understanding the parameters affecting sorption of radionuclides, it is important that the factors controlling desorption of the ions be considered.

A 250-g sample of the air-dried shale was allowed to stand in contact with 2 l. of 1 M NaCl for a period of 72 hr. The suspension was then transferred to a Buechner funnel and leached with an additional 3 l. of 1 M NaCl. Following this treatment the shale was elutriated with ethanol (95%) until a negative test for chloride was obtained. The sodium-treated shale was then transferred to 1 l. of 0.01 M $CsNO_3$ solution tagged with Cs^{137} and allowed to come to equilibrium. The excess $CsNO_3$ solution was removed from the shale by filtration on a Buechner funnel, and the shale was oven-dried. By comparing the activity of the filtrate to the activity of the initial $CsNO_3$ solution, it was estimated that the shale had retained 3.75 mEquiv Cs/100 g. Interstitially retained, cesium would have contributed less than 0.2 per cent of this total even if the excess solution had not been removed by filtration. It is also necessary to point out that the 3.75 mEquiv Cs/100 g retained by the shale in this particular study is about 700 times the partial cesium capacity estimated for the 0.23 M $NaNO_3$ or about 1600 times that estimated for the 0.5 M $NaNO_3$ containing 0.55×10^{-7} M $CsNO_3$.

Six-gram portions of the oven-dried Cs–Na-shale were weighed and transferred to Buechner funnels. Various leaching solutions were passed through the shale samples in 100-ml increments, and aliquots removed for assay of cesium activity. After completion of a leaching series the shale was oven-dried and an aliquot of each shale sample removed for counting. This provided a check on the amount of cesium remaining on the shale.

Ethanol (95%) and distilled water were less effective than 0.01 N NaCl in the removal of cesium from the shale (Fig. 5). Most of the cesium removed by the ethanol and distilled water was probably due to hydrolysis. The hydrolyzed cesium constitutes only a small fraction of the total cesium removed from the shale by salt solutions, as 0.01 M NaCl was much more effective in removing cesium from the shale. The sodium ions were effective in replacing a portion of the exchangeable cesium held by the shale.

A leaching solution was prepared which contained approximately the same stable element composition as the intermediate-level waste solution. The pH, originally 11, was adjusted using concentrated HCl. Portions of the Cs–Na-shale were leached with solutions having pH 11, 10, 8, 4 and 1.8. The amount of cesium desorbed from 6 g of shale was approximately the same for all solutions until the pH was reduced to 1.8 (Fig. 6). The increased cesium removal at pH 1.8 can be attributed to an increase in the ionic activity of the leaching solution and not to a continuous pH effect.

Six-gram portions of the Cs–Na-shale were also leached with solutions of various cations. First, the shale was leached with a liter of the 0.01 N solution in 100-ml increments. This was followed by nine 100-ml increments of 0.1 N solution and nine of 1.0 N solution. Fig. 7 shows a typical leaching curve. The slope of the linear portion of the curve is a measure of the relative effectiveness of the leaching cation. Extrapolation of the linear portion of the curve yields a measure of the cesium that is not easily replaceable but is "fixed".

The nature of the cation of the leaching solution was found to be much more important than the nature of the anion in effecting removal of cesium

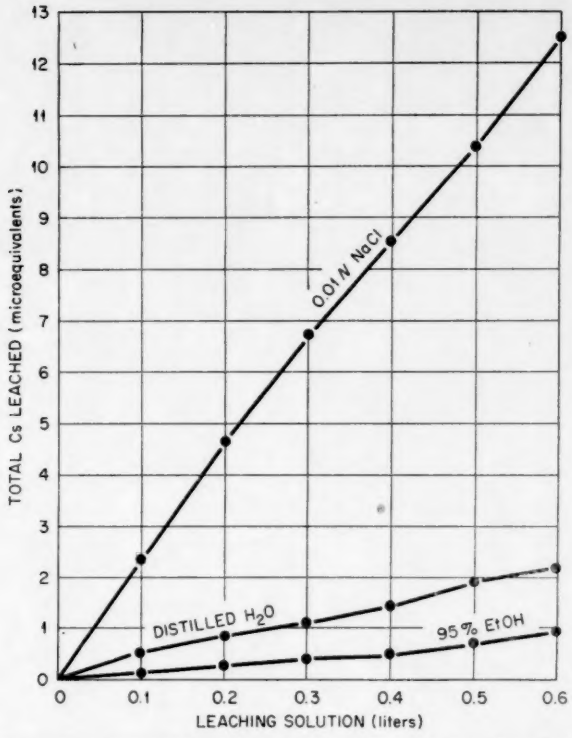

FIG. 5. The effect of exchangeable cations on the removal of cesium from Conasauga shale.

from the shale. Table 3 compares the various cations employed in order of effectiveness of cesium removal with sodium assigned a relative value of 1.0. This series shows a marked dependence on the ionic size and the valence of the replacing cation. The equilibrium studies also indicated these relationships.

Table 3 also shows the amount of cesium that is not easily exchangeable by each of the leaching solutions. The mean value is 1.33 $\pm$ 0.09 mEquiv/100 g shale.

DISCUSSION

The results indicate that of the solution parameters studied, the concentration and nature of cations in the solution are of major importance in affecting retention of cesium by Conasauga shale. The solution pH is of importance only when the hydrogen ion concentration is high enough to change the ionic activity of the solution. The nature of the anion in the solution is of secondary importance.

The most important factor controlling cesium retention is not a solution parameter, but the mechanism by which the cesium is sorbed by the shale. Of the 16.5 mEquiv/100 g cation exchange capacity of the shale, 4.0 mEquiv/100 g or 24 per cent of the exchange sites were not accessible to methylene blue, thus presumably residing within the interlayer spacing of the nonexpanding clay minerals. This value compares favorably with the value of 40 per cent of the exchange sites occurring within the interlayer spacing in illite and suggests that other exchange materials are present in the sample.[5]

Desorption of cesium from the shale indicates that 1.33 mEquiv/100 g is "fixed" or not easily replaceable from the shale. The partial cesium capacity of the shale for cesium from the ORNL intermediate-level waste stream is approximately 0.0056 mEquiv/100 g. Thus, the ability of the shale to retain cesium in a form that is not easily replaceable (1.33 mEquiv/100 g)

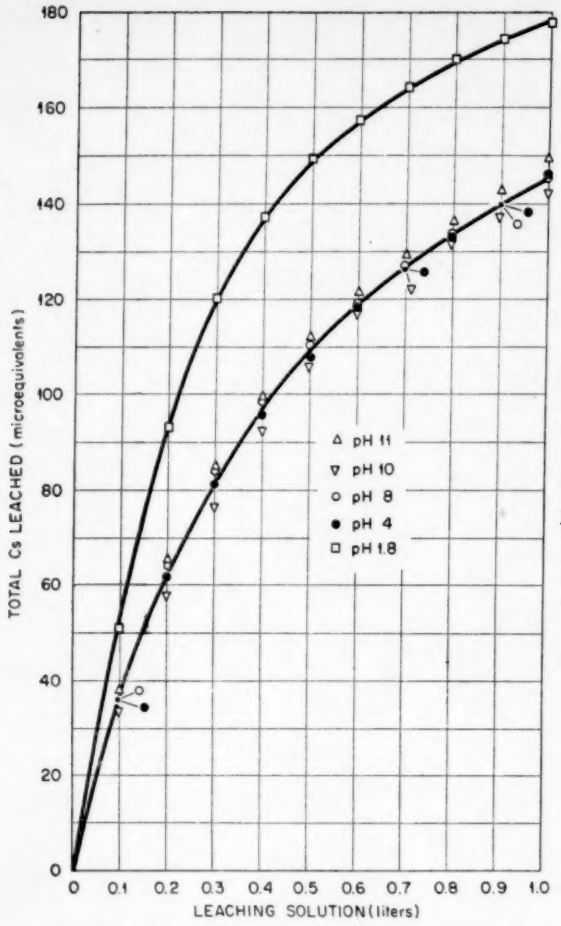

FIG. 6. The effect of pH on the desorption of cesium from Conasauga shale.

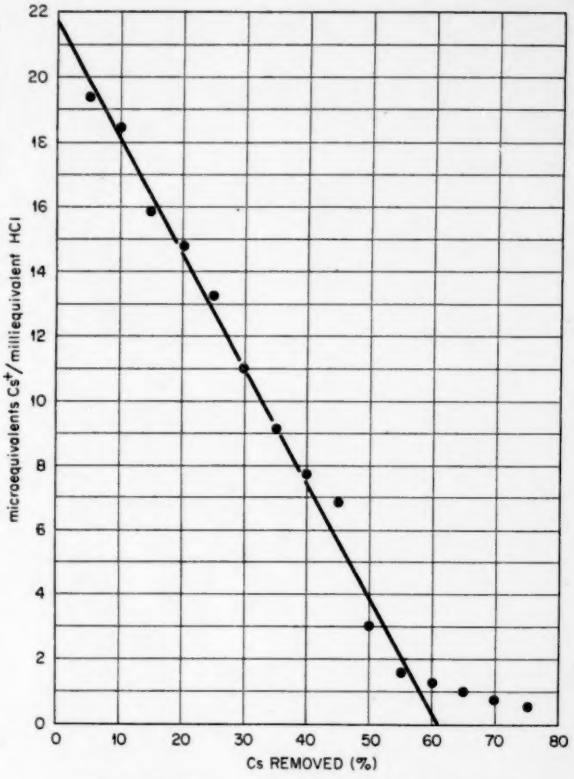

FIG. 7. The desorption of cesium from Conasauga shale by HCl.

greatly exceeds the partial cesium capacity of the shale for cesium from the ORNL intermediate-level waste stream. All of the cesium sorbed from this waste stream would thus be expected to occupy "fixation sites," and the ease of cesium desorption would be much less than that observed in the laboratory experiment. A discussion of the mechanism of sorption of cesium may be found in a separate paper and will not be discussed in detail here.[6]

SUMMARY

Conasauga shale, from the area in which the ORNL intermediate-level waste pits are excavated, has a high affinity for cations of a size comparable to cesium. The fixation of cesium by the illitic Conasauga shale is evidenced by the difficulty of replacing the final 1.33 mEquiv cesium from 100 g shale.

Laboratory results indicate that the fixation capacity of Conasauga shale for cesium would not be exceeded by the partial cesium capacity for the ORNL intermediate-level waste stream. Cesium-137 occupying fixation sites on the shale would be extremely difficult to replace by percolating ground water.

Rain water percolating through the formation would remove the interstitial cesium and only a

Table 3. The relative effectiveness of various cations for the removal of cesium from Conasauga shale

Cation	Relative effectiveness of cesium removal	Fixed cesium (mEquiv/100 g)
Cs^+	22	1.43
Rb^+	17	1.42
H_3O^+	13	1.46
NH_4^+	10	1.17
Al^{3+}	10	1.39
Ca^{2+}	9	1.31
K^+	7	1.15
Na^+	1	1.34
Mean value	—	1.33 ± 0.09

very small amount of the exchangeable cesium. The cesium leached would be resorbed from the leaching solution when unsaturated shale was contacted because of the high affinity of the shale for cesium; thus if the pits are removed from operation before complete saturation of the formation is accomplished, there would seem to be little danger of cesium transport due to leaching processes.

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AUTOMATIC NUCLEAR EMULSION SCANNER

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Abstract—An automatic nuclear emulsion scanner has been designed and constructed to scan the exposed film in personnel radiation monitoring badges. The scanner detects a portion of proton-recoil tracks in the film emulsion caused by incident neutron flux. The scanning speed, at present, is comparable with that of the visual reader and provides about the same statistics.

The scanner uses a special closed-circuit TV camera system to view, scan and count the proton-recoil tracks in a magnified image of a dark-field illuminated nuclear emulsion. The camera video signals are analyzed in a computer that uses two ultrasonic delay lines for signal storage. The computer discriminates against single grains, clumped grains and unusual artifacts in the emulsion. Tracks are detected on the basis of their contrast against their background, and their projected length in the emulsion.

Field evaluation tests at Brookhaven National Laboratory (BNL) indicate that the track count is a linear function of neutron flux. The spectral response curve of the scanner shows a proportionately greater track count at the higher energy levels. In addition, the scanner track count is fail-safe for moderate γ-exposures.

Operational characteristics of the automatic nuclear emulsion scanner are discussed.

I. INTRODUCTION

AN ENGINEERING model of the automatic nuclear emulsion scanner (Fig. 1), described in a paper[1] given at the 1957 National Convention of the IRE, is now being used (at Brookhaven National Laboratory) to count the proton-recoil tracks found in exposed neutron-monitoring film-badges.

The automatic nuclear emulsion scanner (Fig. 1) is designed to detect, with the exclusion of background, the relatively short proton-recoil tracks found in badge-film emulsions.

Since the automatic scanner is able to repeat and check all operations and is not subject to tiring, as a human reader is, it can provide optimum reliability.

This paper will discuss: (1) the present development status of the scanner and (2) the result of the recent performance tests. These performance tests were designed to evaluate: (1) the scanner's response to integrated flux, (2) the track-count sensitivity as a function of incident neutron energy, and (3) the effects of background γ-radiation.

II. THE PROBLEM

The processed NTA (Eastman Kodak) emulsion is about 20 μ thick. Each developed grain is about 0.7 μ in diameter. Most of the grains are randomly distributed. Tracks vary considerably in length; the minimum length that is considered a valid track is about 3 μ. The tracks may occur at any depth of the emulsion and are generally randomly oriented in both pitch and azimuth. They usually have a characteristic fanning-out appearance as they pass out of the plane of best focus. The grains in the track are for the most part aligned and nearly contiguous. However, in the tracks of very fast recoils there may be gaps from $\frac{1}{2}$ to 1 μ wide or greater.

The characteristics of a track that are most easily identified by a two-dimensional scanner are: (1) thinness and straightness along at least

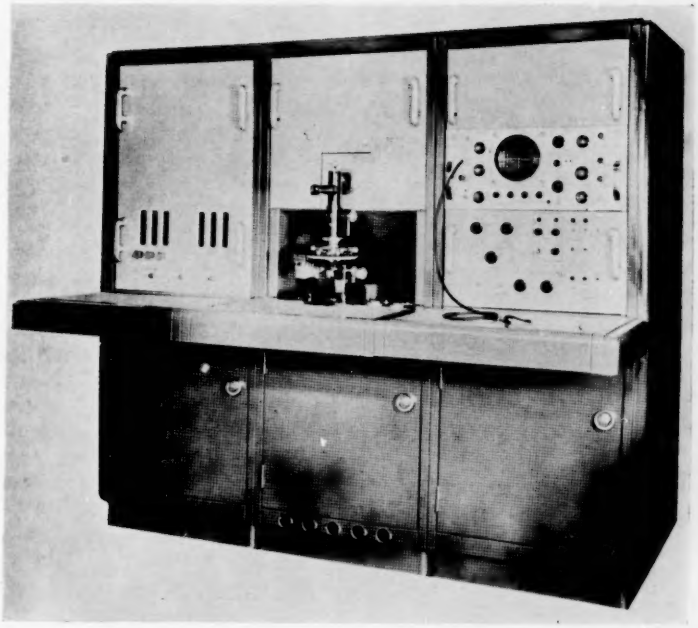

FIG. 1. Automatic nuclear emulsion scanner.

a small portion of its projected length, and (2) the contiguity of its grains. These features are contrasted against those of the background events, which greatly outnumber the tracks. The background includes single grains, clusters of grains, and a large variety of artifacts.

The automatic nuclear emulsion scanner differentiates the thin proton-recoil tracks, which have a minimum projected length of 3 μ or greater, from the other various shapes of granular structures in the emulsion.

Manual methods of neutron dosimetry involve the counting of the microscopic proton-recoil tracks in twenty-five standard fields of 150-μ diameter. The stage of the microscope, with the emulsion mounted on it, is systematically advanced by hand after each field is focused through the entire thickness of the emulsion. The total count of tracks, regardless of length, in the twenty-five fields is then used as a measure of personnel exposure. The maximum permissible weekly exposure (MPE) is 1.2 tracks per field.

III. THEORY OF OPERATION

The exposed film is placed on the stage of a microscope. The image is magnified and directed by means of a right-angle prism to the face of a camera tube. The camera tube converts the light energy of the image into electrical signals. These signals are analyzed in a three-scan-line comparison system that requires the projected length of a track on a horizontal scan-line to be continuous and have a duration corresponding to the minimum acceptable track length. In addition, there must not be any detectable signals, on the preceding line, directly above the detected portion of the track nor, on the succeeding line, directly below the detected portion of the track—that is, there must not be any detectable signals exactly one horizontal scan-line period removed from any point of the detected portion of the track. Thus, extended clumps of grains or artifacts with projected chords greater than the minimum track length are discarded.

This can be described visually using Fig. 2. The video signal from line A is delayed exactly one scan-line period and, in the comparison circuit, the video signal from line B is subtracted from it. The negative video that exceeds a specified threshold is selected and its amplitude is standardized; it is delayed one scan-line period and applied to the coincidence circuit. Similarly, the positive video that exceeds a specified threshold—the remainder when line C is subtracted from line B—is selected, its amplitude is standardized, and it is applied to the coincidence circuit but without delay. In this way the thinness and isolation of the track are established. The continuous length of a track is determined by measuring the duration of the coincidence-circuit output pulse in the track gate.

Fig. 3 illustrates the rejection of a granular structure. The projected image extends across two consecutive scan-lines.

The video from line A is delayed one scan-line period and the video signal from line B is subtracted from it. The negative video that exceeds a specified threshold is selected and its amplitude is standardized. It is delayed one scan-line period and applied to the coincidence circuit. The video signal from line B is delayed one scan-line period, and the video signal from line C is subtracted from it in the comparison circuit. Since the output signal does not, in this

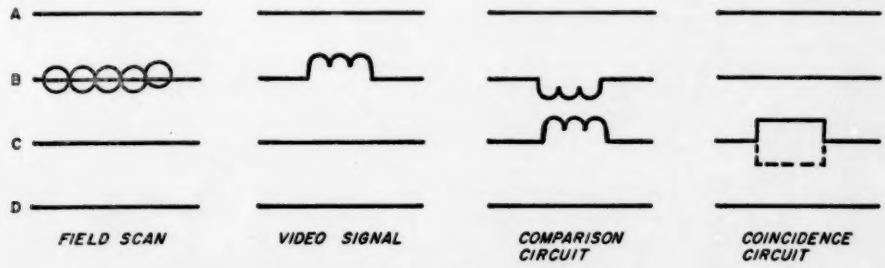

FIG. 2. Track detection (ideal track).

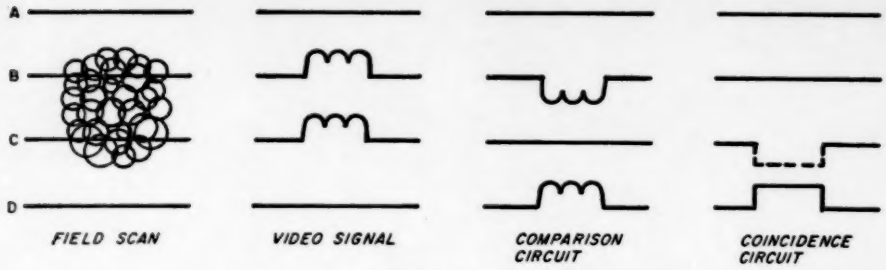

FIG. 3. Track detection (extended cluster).

case, exceed a specified threshold, a signal is not applied to the coincidence circuit, and the lack of thinness of the clump is established. Therefore, a signal is not sent to the track gate to be counted.

The tracks need not be exactly aligned with a raster scan-line nor precisely in focus to be identified by this system. Tests have shown that track angles of from 15 to 20° from the scan-lines will be accepted.

To prevent the over- or under-counting of any predominant track orientation in the emulsion, the film, mounted on the microscope stage, is automatically scanned and moved to present a new area by rotating the stage on an axis that is offset from the optical axis. The stage is rotated through 360° in 2° steps.

Photographs of the monitor oscilloscope of the automatic scanner illustrating the functioning of the track recognition system are shown in Fig. 4.

IV. INSTRUMENTATION

A cardioid dark-field condenser to provide optimum illumination is mounted on the microscope sub-stage and focuses light from a 100-W tungsten lamp in the plane of the film emulsion. The object is magnified by a 40× 1.0 N.A. fluorite oil-immersion objective. The magnification at the face of the camera is about 60×. This provides a square field of about 180 μ on edge when scanned by the camera.

Films to be scanned are held in place with a frame-type clamp designed to hold the film securely to the microscope stage. The working distance between the film and objective is maintained constant by a spacer that is adapted to the objective and held in contact with the film by the weight of the upper structure of the microscope. The vertical axis about which the microscope stage rotates is eccentric with the central ray of the optical path of the system. Thus, as the microscope stage is stepped an entirely new area is illuminated. A complete rotation of the stage traces out an annular ring of illuminated fields having the axis of the stage as the center. The image is directed by means of a right-angle prism to the face of the camera tube for scanner operation. A beam splitter has been inserted in the optical path of the image to provide for operator monitoring of the microscope field.

The camera uses a Vidicon camera tube to convert the optical image of the magnified field to electrical signals. Since the stepping of the microscope stage applies an entirely new image to the face of the camera tube every time the stage is moved, time is allowed for the decay of the old signal and build-up of a new one.

The magnified image of the field is projected onto the face of the Vidicon and scanned. Fig. 5 is a block diagram of the system. Video signals from the camera are fed to line driver (1). The output from line driver (1) is sent through two channels. The delayed channel consists of quartz delay line (1) and delayed channel amplifier (1). This delays the signal for one horizontal scan-line period. The undelayed channel consists of an attenuator, to simulate the attenuation in the delayed channel, and the undelayed channel amplifier, which is identical with delayed channel amplifier (1). The undelayed channel thus simulates the delayed channel in all respects except for the delay. The detected video signals from the two channels are fed into a comparison network. Their difference is amplified in a gain-stabilized video-amplifier. The portion of the system from line driver (1) to the output of the video amplifier is called the canceler, because it accurately cancels (with 98 per cent rejection) any two equal signals appearing at corresponding points on two successive scan lines. Equality of gain in the two channels, to provide accurate cancellation, is maintained by the autobalance circuit.

The output of the canceler is bipolar; both the positive and negative output signals exceeding preset amplitudes actuate the respective Schmitt amplitude

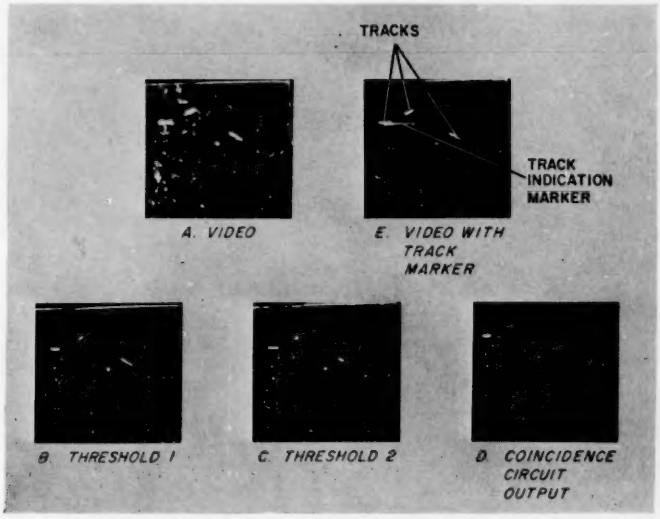

FIG. 4. Photographs of monitor oscilloscope.

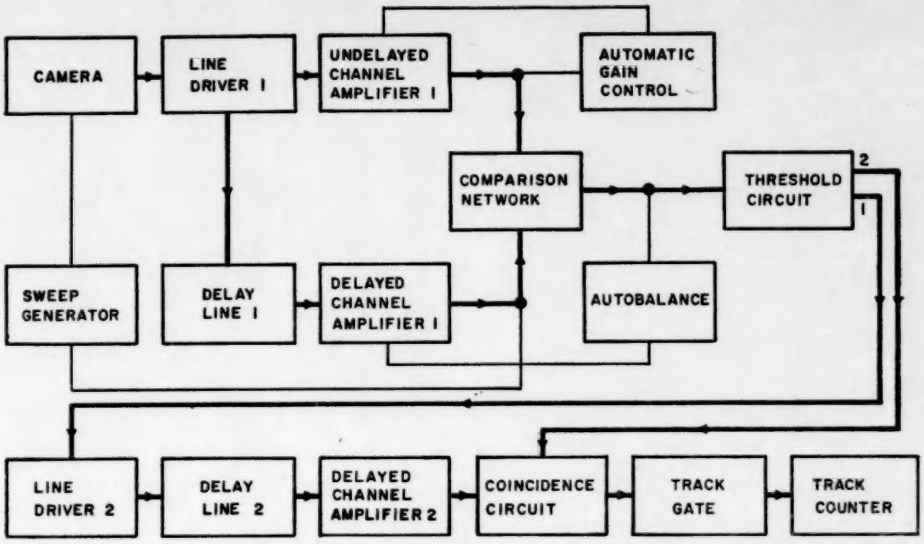

FIG. 5. Block diagram of the scanner.

detectors in the threshold circuits. The outputs of the threshold circuits are fed to a coincidence circuit by two separate paths. The output of threshold (1) circuit (negative input) is delayed one scan-line period in delayed channel (2). Delayed channel (2) consists of line driver (2), delay line (2) and delayed channel amplifier (2). The output of the threshold (2) circuit is applied directly to the coincidence circuit.

Signals that appear simultaneously at the inputs to the coincidence circuit are passed to the track gate. The track gate passes only those coincident signals that exceed a preset duration that corresponds to the minimum track length. The output of the track gate is sent to the track counter.

V. EVALUATION TESTS

Since the early part of 1960, scanner evaluation tests have been conducted at Brookhaven National Laboratory in the Health Physics Division. The Health Physics staff have most helpfully provided the wide variety and range of test exposures which enabled us to evaluate the scanner properly.

One series of tests was designed to evaluate the linearity of scanner track-counts as a function of integrated flux (Fig. 6)*. In these tests a Pu–Be neutron source was used to provide exposures up to 22×10^6 n/cm^2. As a basis for comparison, the visual readings of these films are shown. The instrument track-count is the reading obtained for 180 fields, or one rotation of the microscope stage. The visual-reader track-count is for twenty-five standard fields. The linearity of track-count as a function of the exposure is good in both cases.

A series of tests was designed to determine the energy response of the scanner. Neutron fluxes with energies of 0.44 to 5.15 MeV and 13.5 to 15 MeV, obtained through the use of the BNL Van de Graaff accelerator, were used as test films. Plots of the instrument and visual counts as a function of neutron energy are shown in Fig. 7. The ordinate is normalized to an exposure of 1 MPE based on the table of 40 hr/week maximum permissible fluxes found in NBS Handbook 53. Scanner efficiency is relatively low at the low energy end of the spectrum. At higher energies the scanner sensitivity varies in a manner similar to that of the visual reader.

Another series of tests was designed to determine the effect of background fogging by γ-radiation on the scanner sensitivity. Fig. 8 shows the instrument count as a function of γ-flux. The increase in indicated track-count

* *Editor's note*: This statement is true for any given spectrum.

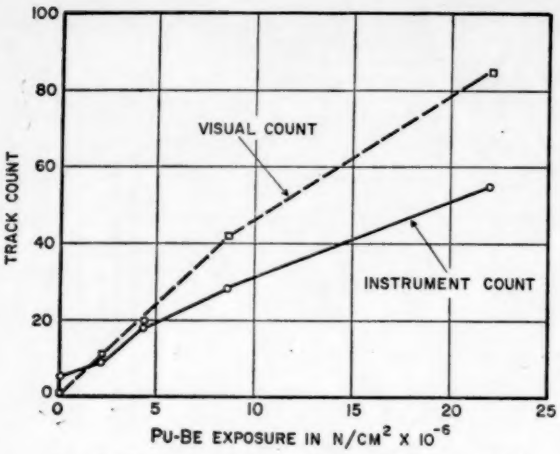

FIG. 6. Track count as a function of integrated flux.

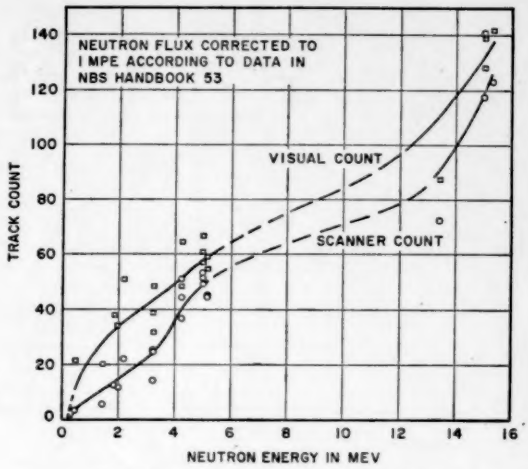

FIG. 7. Spectral sensitivity of scanner.

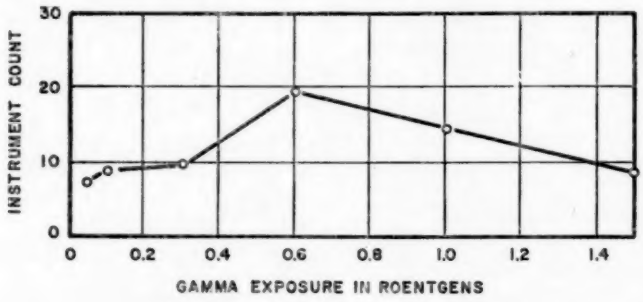

FIG. 8. Scanner response to γ-fogging.

up to 600 mr is caused by the increased combinations of random grains which, when projected in two dimensions, appear like tracks. Above 600 mr the increasing number of developed grains apparently increases the background illumination to the point where grain contrast deteriorates. This causes the track-count to diminish. The results obtained by scanning films exposed to Pu–Be neutron fluxes and three levels of gamma radiation are shown in Fig. 9. The scanner's sensitivity decreases with increasing γ-flux, but the count contributed by the gamma flux provides a fail-safe indication for exposures up to approximately 8.64×10^6 n/cm^2 which is the equivalent of 1 MPE.

VI. FILM SCANNING

The procedure used for scanning personnel neutron monitoring films in the developmental scanner is not fully automated. Mounting of the film, which includes oiling below and above the film, and lowering the microscope draw tube are manual operations. After the film and microscope are in position, the operator presses a button. This starts the automatic scanning of the film and is accomplished in about 45 sec. The scanning time required is independent of the film exposure. When the scan is completed, decade counters indicate the track count for the film. The recording of these data is manual. All of the manual procedures, including electronic checking of the equipment, required an average of 2.5 min per film over about a 2-hr period.

A detailed analysis of the manual scanning time has not been made. It is estimated, however, that the time to scan and record films with very low exposures is approximately 2.5 min. When exposures of about 5 MPE are scanned,

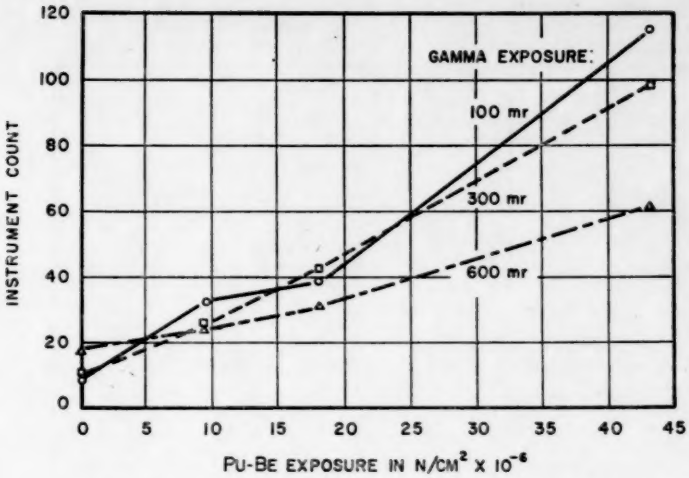

FIG. 9. Scanner response to integrated flux in a γ-background.

reading times of up to 10 min can be expected. In addition, operator fatigue becomes a factor after several such films are read.

VII. OPERATIONS

Because the scanner has been used only for a few months, an evaluation of its operation is necessarily limited. However, some of its operational characteristics can be discussed.

The instrument requires a more thorough calibration procedure than other types of film readers—for example, beta–gamma densitometers, because of the more sophisticated circuitry that is required. Since the scanner is sensitive to image contrast and image shape, the threshold and track gate circuits require careful adjustment. These circuits are adjusted with the aid of electronic test signals sent through the instrument at the input to the camera video amplifier. Test-signal levels, and the functioning points of the threshold circuits are monitored on the test oscilloscope. The track gate circuit response is also monitored to determine operation when acceptable and nonacceptable test-signal widths are applied. The monitor oscilloscope is used to provide this check. After these adjustments are completed, operation for several hours without readjustment is possible. The procedure has been to adjust the electronic operating levels of the instrument before the morning and afternoon periods of operation, and to check them afterward.

Adjustment of the illumination and optics of the system require careful attention. Illumination is monitored by a photocell mounted on the illuminator, and only the illuminator bulb brilliance is monitored. The operator must use his trained eye to determine whether the object illumination is uniform and "best". As a check a test film with a known exposure is scanned. If the instrument does not give a correct reading, the illumination and optics require adjustment. When these have been correctly adjusted, they are stable for several hours. This calibration is also checked before and after long periods of operation by scanning a known test film. Because immersion oil is used between the microscope objective and the film, repeated agitation of the oil by the motion of the microscope stage causes the formation of minute bubbles around the edges of the objective. As their numbers increase they have a deleterious effect on the dark-field illumination. These bubbles are troublesome after about ten films have been scanned. It has, therefore, been the practice to clean and re-oil the objective after ten films have been scanned.

No data on the variance in total track count for the automatic scanner are available. The variance is clearly a function of the condition of

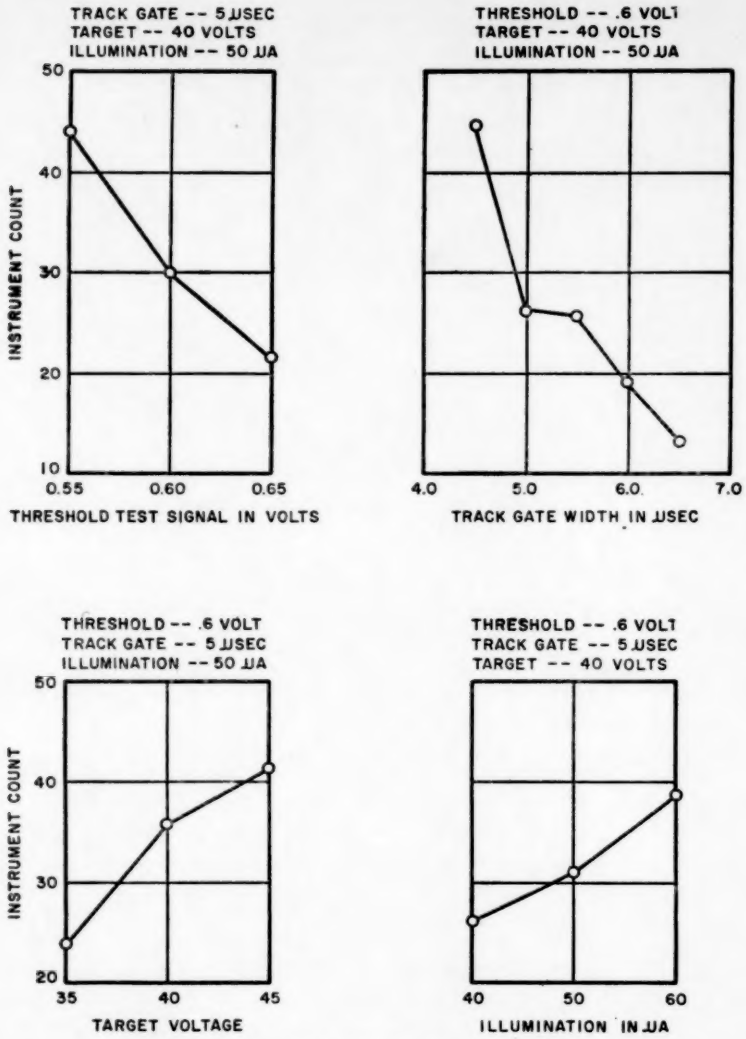

FIG. 10. Control sensitivity of automatic nuclear emulsion scanner.

the film which is scanned. A typical scanner track count is shown in Fig. 10 as a function of the system parameters.

The amount of maintenance required for scanner operation has proved satisfactory. In the first 350 hr of operation at Brookhaven National Laboratory, two faulty plugs were discovered, five tubes failed, and one short circuit occurred. These failures resulted in a total of six interruptions over a period of about $4\frac{1}{2}$ months.

VIII. CONCLUSIONS

The visual method of scanning is somewhat faster than the present automatic scanner where there is little or no exposure on a film. However, for films with a considerable amount of exposure, the automatic scanner is faster than the visual scanner and does not have the side effects that the visual scanner has. The automatic scanner count can easily be repeated and checked against the count of a calibrated test exposure. The automatic scanner is less susceptible to

reading errors caused by background fogging. The spectral response of the automatic scanner is comparable with that of the visual reader except at low energies. Operationally, the automatic scanner requires a technician well trained in the handling of optics and in electronic adjustments.

Acknowledgments—The writer wishes to take this opportunity to acknowledge the contributions to the work on this project by Messrs. K. C. Speh of AIL, F. P. Cowan, F. Lacey, F. Rizzo, of the Brookhaven National Laboratory Health Physics Division, and the girls in the Personnel Monitoring group at Brookhaven, who tirelessly read all the necessary test exposures.

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1. S. Becker and J. B. Franceschini, *Track Recognition System for Scanning Nuclear Emulsions*. IRE National Convention Record, Pt. 9, p. 46 (1957).

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NOTES

Automatic α-Particle Hand Counter Model HC-2

(*Received* 1 *June* 1960)

1. Introduction

THE HC-2 is a completely automatic, personnel-operated, α-particle hand monitor. It is a qualitative instrument telling the person using it only whether his hands are "cold" (under a predetermined level) or "hot" (over that level). A person simply walks up to the counter, presses the "start" button, then holds his hands under the probes until either the "safe" lamp or the "hot" lamp comes on. The machine then resets itself automatically. The time required for counting is set into the instrument and may be anywhere between 3 and 40 sec. The dividing line between hot and cold is also adjustable and is set according to the particular application.

The HC-2 is a.c. powered and partially transistorized. The low-voltage supplies are all selenium-rectifier operated. The high-voltage supply is a transistor oscillator type with output voltage variable from 2700 to 3000 V. All timing is done electronically. All components are mounted on a standard $8\frac{3}{4}$-in. × 19-in. panel chassis attached to an $8\frac{3}{4}$-in. × 19-in. relay rack panel, shown in Fig. 1. The electronics chassis and panel have been designed to fit into all hand counter systems now in existence at Rocky Flats and replace the present electronic chassis. The probes used are Mylar-covered, four-parallel-wire air proportional probes of Rocky Flats design. They have an operating potential of from 2600 to 2700 V. Each counter has two probes mounted either on top of the cabinet or underneath the chassis, depending on the particular model.

2. Operation

The "ready" light will be on whenever the HC-2 has reset itself and is ready for use. While on "ready" the count rate accumulating and timing circuits are de-energized. Thus, the background count of the instrument does not accumulate during idle periods. When the "start" button is depressed, the "ready" light goes off and the "count" light comes on. As long as the button is held depressed, nothing further happens. When the button is released, two things happen: the count rate accumulating circuit is energized and the timing cycle starts. If, at the end of the timing cycle, no counts have entered the count rate circuit, the "safe" lamp will come on as the "count" lamp goes off. The "safe" lamp stays on long enough for its condition to be observed, then the reset timer fires, resetting all circuits and returning the instrument to the "ready" condition. During the "count" time, if enough counts enter to trigger the "hot" circuit, the "hot" lamp will come on instead of the "safe" lamp. It will remain on as did the "safe" lamp, then reset back to "ready". It should be noted that the "hot" lamp will be energized as soon as enough counts have arrived, which means that the higher the count rate the sooner the lamp will come on. As a safety precaution, the "hot" and "safe" circuits are so designed that both lamps cannot be on at the same time.

The hot circuit will not trigger at all for input pulse rates less than one pulse per sec. However, at rates above this, the point at which the hot circuit will trigger is determined by the number of pulses that arrive, regardless of the time in which they arrive. If the sensitivity is set at six pulses, say, then as soon as six pulses have been received the circuit will trigger. If the rate of arrival of these six pulses is two per sec, then the circuit will trigger in 3 sec. If the arrival rate is reduced to one per sec, the circuit will trigger after 6 sec have elapsed. When the incoming pulses are randomly spaced, which is the case, then the triggering level or time becomes a variable governed by the statistical deviation of the count rate. The effect of this fluctuation is illustrated in Table 1 which shows the minimum "hot" triggering point one may expect for various counting times and circuit sensitivities. The range given for any particular setting is that due to the standard deviation spread. The values given are based on a hand counter probe efficiency of 30 per cent.

From the above it can be seen that there will be a variation in the "hot" triggering point from one count to the next. When setting the instrument, the count time should be set first for as long a time as is practical. The range (or sensitivity) should then be set so that the "hot" circuit will trigger at least 80 per cent of the time at the desired count rate. The statistical deviation shown in Table 1 will probably not allow a much better accuracy figure than this for rates under 1000 counts/min. As the table shows, the

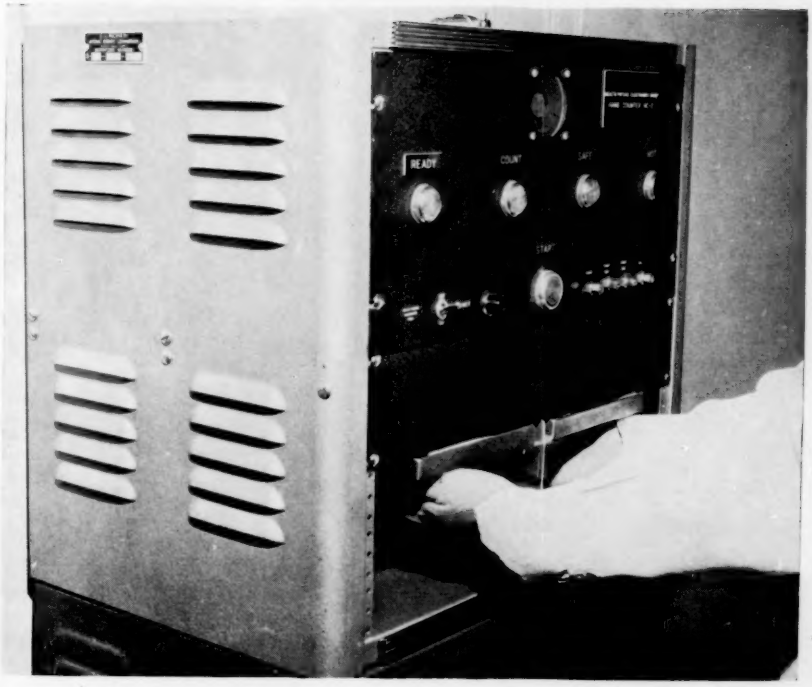

FIG. 1. Hand counter.

Table 1. *Count rate range at the probe necessary to trigger the "hot" lamp, in counts per minute*
(Based on standard deviation)

Counting time (sec)	Sensitivity (no. of pulses)						
	3	4	5	6	7	8	9
3	84 to 320	135 to 400	185 to 480	240 to 560	295 to 640	350 to 720	400 to 800
4	84 to 320	100 to 300	135 to 360	180 to 420	220 to 480	260 to 540	300 to 600
5	84 to 320	100 to 300	110 to 290	145 to 340	175 to 385	210 to 430	240 to 500
6	84 to 320	100 to 300	110 to 290	120 to 280	145 to 320	170 to 360	200 to 400
7	84 to 320	100 to 300	110 to 290	120 to 280	125 to 275	145 to 310	170 to 340
8	84 to 320	100 to 300	110 to 290	120 to 280	125 to 275	130 to 270	150 to 300
9	84 to 320	100 to 300	110 to 290	120 to 280	125 to 275	130 to 270	135 to 265

$$\text{Pulses per sec at trigger input} = \frac{\text{counts per min at probe}}{180}$$

triggering accuracy gets worse as the count rate becomes lower.

The length of time that either the "safe" or "hot" lamp remains on is adjustable (front panel control) from 1 to 6 sec. This hold time need be set just long enough so that a person can note his reading, then step aside for the next person in line. From 2 to 3 sec should be adequate for this.

A loudspeaker with adjustable volume is included where an aural indication of count rate is desirable.

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Improvements in Personnel-Metering Procedures at the National Reactor Testing Station

(*Received* 1 *April* 1960)

The Personnel Metering Branch of the Health and Safety Division, Idaho Operations Office (IDO), began the use of a modified version of the Hanford-type plastic film badge at the National Reactor Testing Station in March of 1958. These badges contain filters of 0.0393 in. Cd, 0.005 in. Ag and 0.0191 in. Al, providing absorber thicknesses of 950 mg/cm^2, 203 mg/cm^2 and 175 mg/cm^2, respectively, including the plastic slide in which they are mounted. The plastic security insert provides an open-window thickness of approximately 100 mg/cm^2. Exposure response experiments using several types of personal dosimeter film packets were performed, and the DuPont Type 558, which is a combination of 508 sensitive film and 1290 insensitive, was selected. The resultant curves of three of these experiments are shown in Fig. 1. Films were exposed to radium gamma at twelve different dose levels ranging from 50 mr through 800 r. Other experiments using various uranium beta and radium gamma combinations at 10–20 and 30 mr levels were also performed. The IDO Instruments and Development Branch had designed and constructed an extremely sensitive automatic film reader. (This was reported at the 1958 HPS meeting.) Using this machine and the DuPont 508 film, exposures of 10, 20 or 30 mr of Ra gamma or U beta are measurable within 12 mr with 95 per cent confidence.

Table 1 shows the response of groups of 5 film to the various exposures. The reader will read and prepare detail cards for 98% of the exposures experienced at the NRTS (zeros and high-energy gamma.)

One of the most important limitations of film dosimetry is the fact that the density does not respond

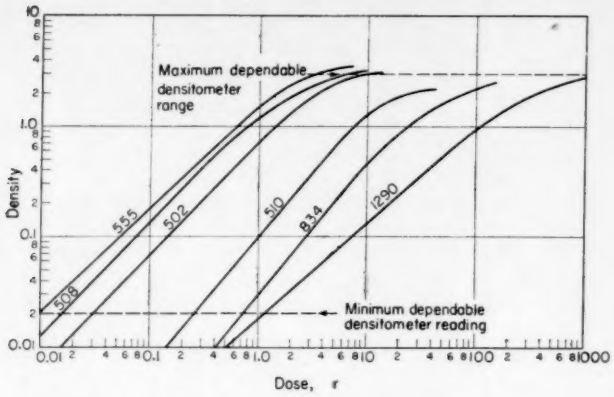

FIG. 1. Ra gamma calibrations. Shield readings (1 mm Cd).

Film packet no.	Film type: Sensitive	Insensitive
552	502	510
558	508	1290
544	555	834

Table 1. *Response of DuPont type* 508 *film to low-level exposures*

Exposure (mrem) U beta	Ra gamma	Machine reading (mrem) Beta	Gamma	Exposure (mrem) U beta	Ra gamma	Machine reading (mrem) Beta	Gamma	Exposure (mrem) U beta	Ra gamma	Machine reading (mrem) Beta	Gamma
		10	12			22	7			27	13
		8	6			22	6			30	12
10	10	5	11	20	10	16	9	30	10	29	11
		10	10			20	7			30	14
		7	11			13	9			25	10
5 Film average		8.0	10.0	5 Film average		18.6	7.6	5 Film average		28.2	12.0
		11	11			19	17			29	13
		9	16			20	19			25	16
10	20	10	16	20	20	18	22	30	20	30	17
		10	13			23	13			32	17
		17	18			20	17			30	16
5 Film average		11.4	14.8	5 Film average		20.0	17.6	5 Film average		29.2	15.8
		9	30			18	25			36	25
		10	29			19	30			30	26
10	30	7	29	20	30	20	23	30	30	30	24
		9	23			19	24			32	26
		16	26			19	30			26	24
5 Film average		10.2	27.4	5 Film average		19.0	26.4	5 Film average		30.8	25.0

linearly with radiation exposure. This may be partially overcome by developing a standard correction curve or chart. By using these standard corrections, it is possible to obtain a density–exposure curve which is linear up to near the film saturation point. Such a correction chart for DuPont Type 508 film has been developed by the Personnel Metering Branch and is used on all film net densities greater than 0.40.

To aid in energy determinations, badges and film were made available to the National Bureau of Standards who performed a series of exposures to low-energy photons and exposed one group to Co^{60}. Uranium beta exposures were added to a portion of these films after their return to IDO. The densities produced behind the various shields, using Dupont 558 packets is shown in Fig. 2.

From a plot of density versus exposure for the various filters (Fig. 3), it is apparent that a film which has received a gamma exposure of energy greater than 25 keV, should fall into one of the following three categories:

(1) 25–55 keV: Al density is much greater than Ag density, and OW density is 0–19 per cent greater than Al density.

(2) 55–85 keV: Al and OW densities are approximately equal and both are greater than the Ag density.

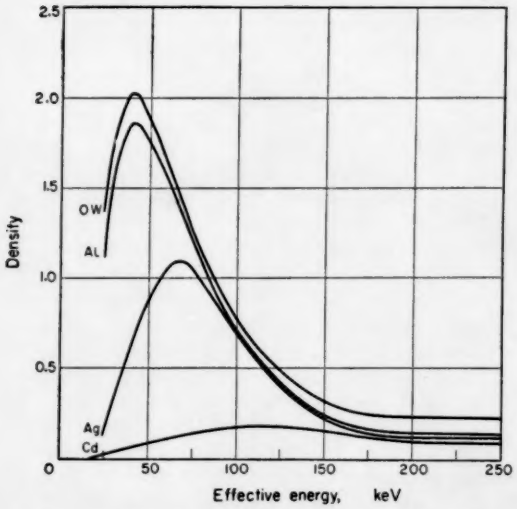

FIG. 2. Relative response of DuPont 508 emulsion with various filters. Exposure — 100 mr gamma +140 mrems uranium beta; open window (~100 mg/cm²); 0.0191 in. Al filter; 0.005 in. Ag filter; 0.0395 in. Cd filter.

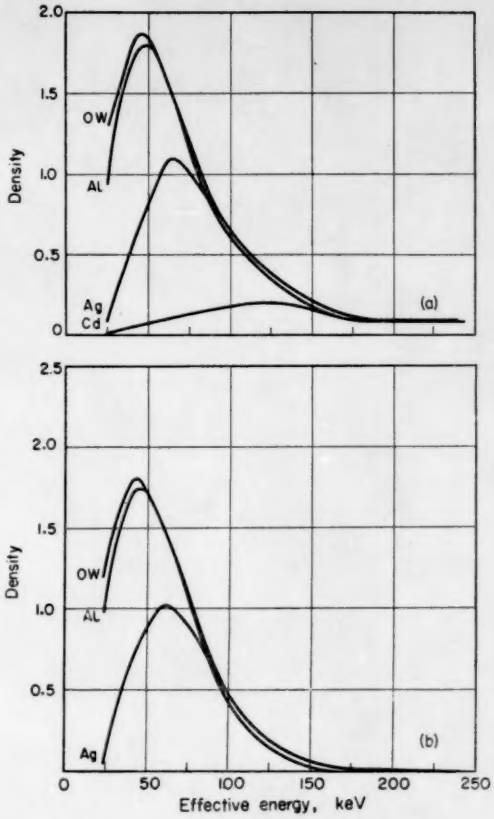

FIG. 3. Relative response of DuPont 508 emulsion with various filters. (a) Exposure— 100 mr gamma; film type—DuPont 508; open window (~100 mg/cm²); 0.0191 in. Al filter; 0.005 in. Ag filter; 0.0393 in. Cd filter. (b) Exposure — 100 mr gamma; film type — DuPont 508; open window—Cd; Al—Cd; Ag—Cd.

(3) Above 85 keV: All three densities are approximately equal.

When a film has been exposed to beta radiation from a uranium slab, it has been experimentally proven that the ratio of the OW to Al to Ag density will be approximately 10:2:1.

By use of a plot of the ratios of the various densities (Fig. 4), and through simple manipulations involving only basic arithmetic, film densities due to beta radiation and those due to low-energy photons may be differentiated. Excellent beta-exposure determinations may be obtained by this method. It is also

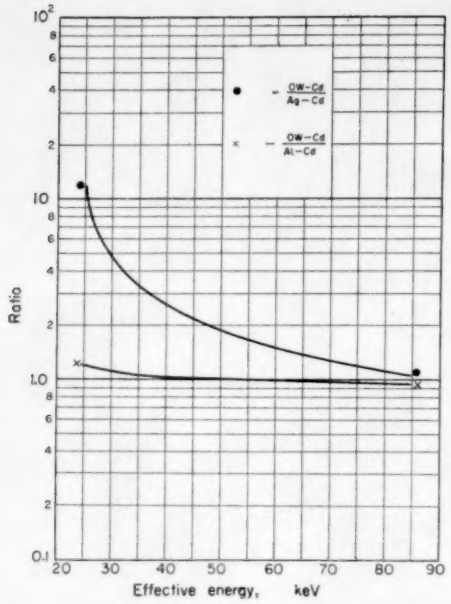

FIG. 4. Variation of ratios with energy.

possible to estimate average photon energies with reasonable accuracy in the range of 25–85 keV.

Dosage determination procedure

(1) Make probit corrections of net densities.
(2) Subtract Cd density from Al, OW and Ag densities. (Corrected)
(3) Inspect remaining densities to determine whether or not they fit either of the three gamma categories or the expected 10:2:1 ratio of a uranium-slab beta exposure. If not, continue to step (4).
(4) Subtract an estimated beta density (10:2:1 ratio) from the Al, OW and Ag densities, leaving densities which fall into one of the above three gamma categories.
(5) Compare the apparent gamma energy indicated by the OW to Al ratio with the apparent energy indicated by the OW to Ag ratio, as shown by graph.
(6) If the two ratios do not indicate the same energy, adjust the estimated beta density and repeat step (3) until the two ratios indicate the same gamma energy.
(7) Convert densities to dose by using standard calibration curves.

This system is simple enough for personnel-metering technicians to perform with little difficulty (Table 2). Other cases require individual handling.

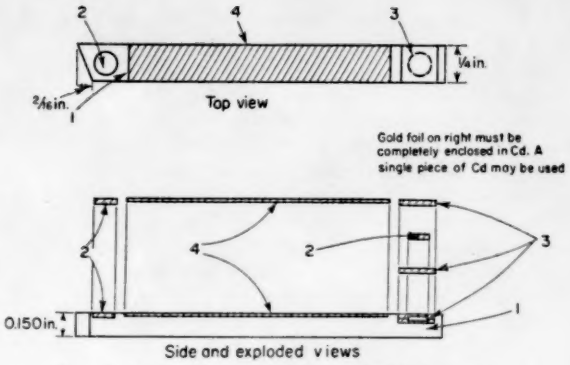
Top view
Gold foil on right must be completely enclosed in Cd. A single piece of Cd may be used
Side and exploded views
0.150in.

FIG. 5. Personnel neutron-threshold detector. (Foils are molded as an integral part of the detector.)

In addition to the above, the badge periods were rearranged so that each area is on a monthly or bi-weekly schedule except for a few where the anticipated weekly exposure is high enough to warrant servicing badges weekly. Utilizing the 508 film, the new film reader and a monthly badge servicing, it is possible to detect and measure exposures of 3–4 mr/week with reasonable accuracy.

Owing to recent criticality incidents, both area-disaster monitoring and personnel-disaster metering are planned. To accomplish the area monitoring, threshold detectors (ORNL type) will be distributed throughout the various areas for determination of the neutron-energy distribution in case of an incident. For the personnel metering we are having inserts fabricated of sulphur, gold, indium and cadmium (Fig. 5) which will be placed into the space obtained by modification of the Hanford-type badge. These will be an integral part of all regular badges. Preliminary calibrations performed with the co-operation of Mr. EDWARD FAST, Materials Testing Reactor, and Mr. EARL EBERSOLE, IDO Analysis Branch, would indicate the ability to detect and measure neutron exposures of approximately 5 rads to personnel.

All information gathered from various personnel-metering devices must be made a matter of record which can be readily produced and made available for preparations of various statistical reports regarding exposure experience. The IDO Personnel Metering Branch has established a data-processing section for this purpose, and is presently equipped with three IBM 026 printing-card punches, one 526 printing summary punch, one 077 collator, one 402 alphabetical accounting machine and one 101 electronic statistical machine.

The master index provides a complete record of all

Table 2. Typical calculation

	Al	OW	Ag	Cd
Net densities	1.12	1.29	0.68	0.48
Step 1. Probit correction	1.24	1.45	0.72	0.50
Step 2. Subtract cadmium density from Al, OW and Ag	0.74	0.95	0.22	
Step 3. Estimate beta density and subtract	0.03	0.15	0.015	
	0.71	0.80	0.205	
Compare apparent energies				
$\frac{OW}{Al} = \frac{0.80}{0.71} = 1.12$—29 keV; $\frac{OW}{Ag} = \frac{0.80}{0.205} = 3.9$—32 keV				
Step 4. Adjust beta density and recalculate	0.74	0.95	0.22	
	0.04	0.20	0.02	
	0.70	0.75	0.20	
$\frac{OW}{Al} = \frac{0.75}{0.70} = 1.07$—33 keV; $\frac{OW}{Ag} = \frac{0.75}{0.20} = 3.75$—33 keV				

Calculated exposure	Actual exposure
Beta density of 0.20 = 400 mrems	400 mrems beta (uranium slab)
Gamma density of 0.50 = 430 mr	400 mr radium gamma
	37 mr 90 kVP X-ray

clearances, badges changes, etc., for every individual at the NRTS, whether utilizing the personnel-metering service or not. It consists of a permanent master file containing one card for each transaction concerning each individual possessing an active badge and showing the previous as well as the current status; the area work decks containing one card for each individual possessing an active badge in an area (these are current status cards only and are used for actual machine work in the data processing); and the contractor work decks containing one card for each individual possessing an active badge during a calendar quarter for each contractor (these are used to prepare quarterly contractor exposure summaries regardless of the area in which the individual was employed, and also to prepare the annual contractor summary report).

Detail cards showing date, badge number, results, etc., are automatically punched as film is read by the automatic reader; neutron results and miscellaneous film results are hand punched by personnel-metering technicians. Periodically, master cards and detail cards are sorted, collated and put through the 402 machine to prepare the personnel-exposure report, personnel-exposure summary or the contractor personnel-exposure summary. In preparing the monthly

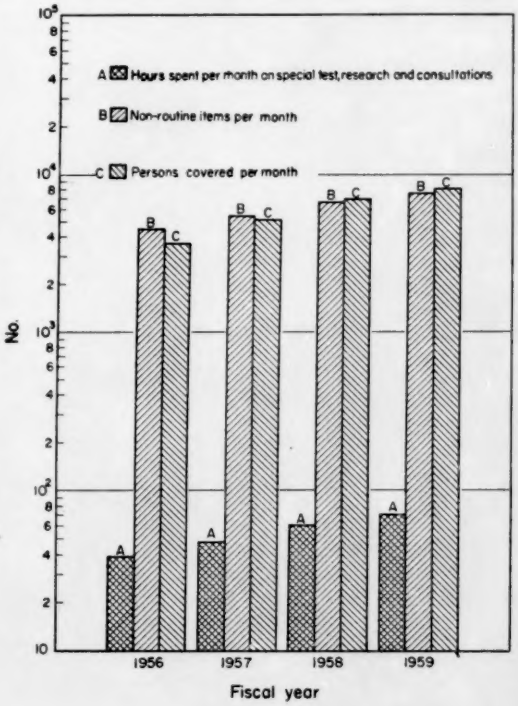

FIG. 6. Increased load.

badge correlation chart and the annual exposure-experience report, the 101 statistical machine is used.

By carefully planning the master card and properly designing the various detail cards, many types of statistical reports are possible.

The steady growth of the NRTS is depicted by this graph (Fig. 6) showing the increase in the three basic categories of the personnel-metering branch. As the NRTS continued to grow, procedures have been streamlined and techniques improved so that the efficiency of the group has increased from a calculated 120 persons covered per technician per month, at the beginning of the program, to 187 persons covered per technician per month, just prior to the change in system in March 1958, and, finally, to 229 persons covered per technician per month, at the end of 1958. This graph (Fig. 7) shows the number of persons covered as of January 1 each year, the number of technicians employed and the number of personnel coverages maintained by each.

The change to DuPont 558 film packets in conjunction with the Hanford-type multiple filter badge and use of the film reader have increased detection and determination capability considerably, and the punched-card data processing has increased the efficiency of record and report generation.

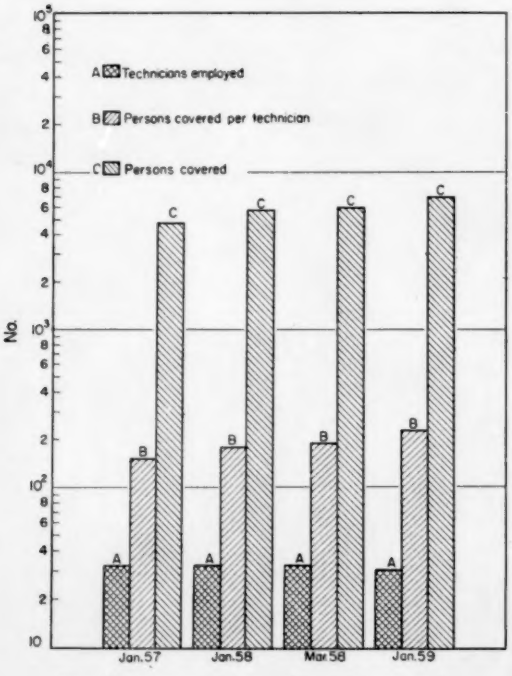

FIG. 7. Personnel-metering performance.

Even greater efficiency is anticipated as personnel gain experience and dexterity in tabulating equipment operation and the programs and procedures are further improved.

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Radiation Glo-Bar—A Device for the Delineation of a Primary X-ray Beam

(*Received* 10 *March* 1960)

DURING the course of radiation inspections and surveys of medical and dental X-ray units, one of the most serious defects found is the lack of proper collimation of the primary beam, causing much unnecessary and undesirable radiation to be delivered to the patient, and also causing increased amounts of scatter radiation to reach the operator. A satisfactory method to demonstrate the state of collimation is the "Radiation Glo-Bar".

The "Radiation Glo-Bar" (see Fig. 1) consists of four arms, each 18 in. long, emanating from a central hub, in the shape of a cross. A strip of fluorescent screen is imbedded throughout the length of each arm. The entire "Glo-Bar" can be attached to a cassette or a table top by means of a suction cup on the hub. Construction is of lucite. The "Glo-Bar" has certain advantages over using a large square fluorescent screen. Firstly, the cost of a single large fluorescent screen monitoring the same size area as the "Glo-Bar" would be greater (36 in^2 of screen for the "Glo-Bar" as against 1296 in^2 of a single sheet screen); secondly, the "Glo-Bar" is more easily transported than an ordinary square screen, the arms of the "Glo-Bar" are removable and the entire device can be carried in a briefcase; thirdly, the "Glo-Bar" screen is more rugged since it is imbedded in lucite; and fourthly, it is more presentable and convenient to attach to vertical areas than the ordinary square screen.

During the radiation inspection or survey of any medical or dental units, the effectiveness of the collimation used by the operator can be vividly demonstrated by the use of the "Glo-Bar" which delineates the primary beam and shows the field being irradiated.

The procedure is as follows:

The "Glo-Bar" is mounted in the position the patient generally occupies during an X-ray examination. The lights in the X-ray room are dimmed

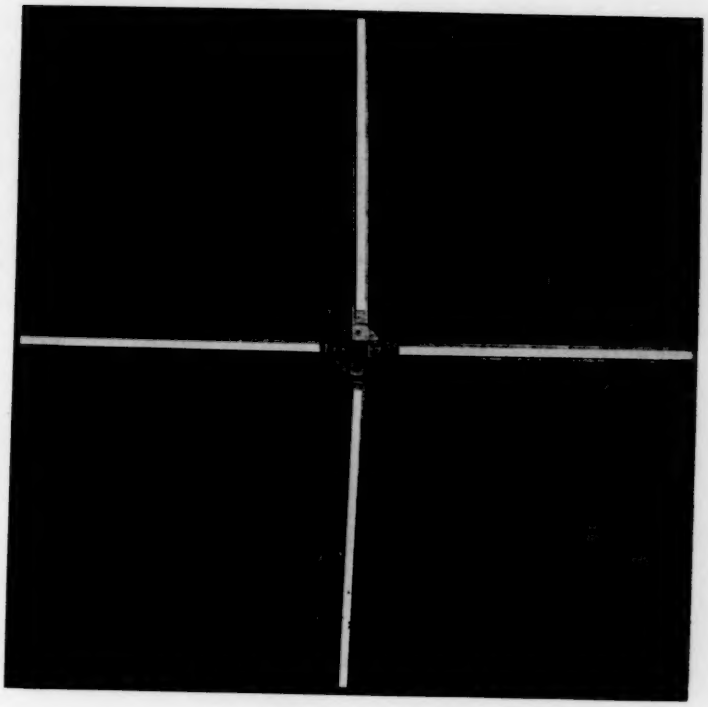

FIG. 1. "Radiation Glo-bar."

(total darkness is not necessary due to the high speed of screens) and an X-ray exposure is made against the "Radiation Glo-Bar". Those portions of the screens not irradiated remain darkened and, therefore, the primary beam on all four sides is sharply delineated. As a further aid in determining the size of the field irradiated, a scale in inches can be scored on the inside of the lucite bars.

In many cases, this is the only way to demonstrate vividly and promptly to operators of X-ray producing units that the area being irradiated is too large and collimation, or better collimation, must be employed.

The use of this device will help to reduce radiation to the individual patient, and, more important from a public health viewpoint, will help to reduce the gonadal dose to the general population as a whole.

At present there are twelve of these devices being used by Radiation Inspectors throughout New York State. The reports of their effectiveness as an inspection and educational tool are enthusiastic.

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LETTER TO THE EDITOR

A Simple Film Badge

(*Received* 13 *January* 1960)

A FILM badge is described, where the radiation filters serve also as support for the film.

Persons exposed to γ-rays or to thermal neutrons wear an appropriate dental film shielded on both sides by one cadmium and one tin filter (1 mm thick). Persons exposed to β-rays or to soft X-rays wear a film shielded on both sides by an aluminium filter of the same thickness.

The shape of the filters is as follows. A strip of cadmium or of tin ($16 \times 32 \times 1$ mm) (Fig. 1) is folded in the middle (Fig. 2), the distance between the inner faces of the strip being 1.2 mm, so that the film can be inserted in the pleat without pressure (Fig. 3). The tin filter is placed beside the cadmium filter as shown in Fig. 3. The aluminium filter is longer ($32 \times 32 \times 1$ mm), but folded in the same manner as described previously [Figs. 4(a) and (b)].

The name of the person wearing the badge is written on an adhesive paper and stuck on the filter. The film with its filters is put into a bag of plastic material (cellophane + polyethylene, 0.1 mm thick). The bag is welded and fixed with one or two safety pins. The badge is thus ready for use (Fig. 5).

These film badges are currently used by us for personnel monitoring. For the calibration of films with the aid of a shielded calibration device[1], we use the same film badge, adding an aluminium filter of the same dimensions as the cadmium filter (Fig. 6). Thus the four portions of the film, unfiltered and filtered by cadmium, tin and aluminium, are exposed simultaneously to the shielded source.

The described, very simple kind of film badge is derived from similar badges, which we had the opportunity to see during a visit to the departments of film dosimetry of CEA in Saclay (France) and of AERE in Harwell (Great Britain).

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Reference

1. V. PAIĆ, *C. R. Acad. Sci., Paris* **249,** 1951 (1959).

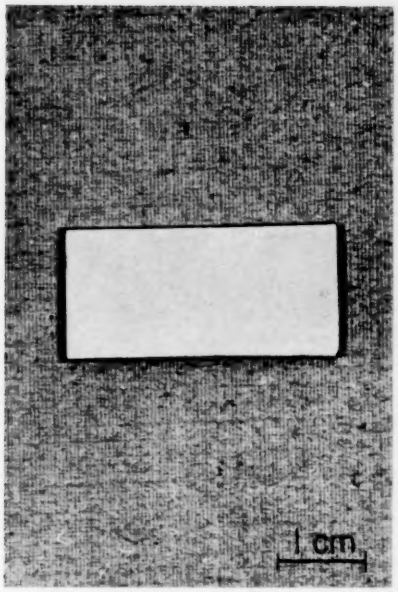

FIG. 1.

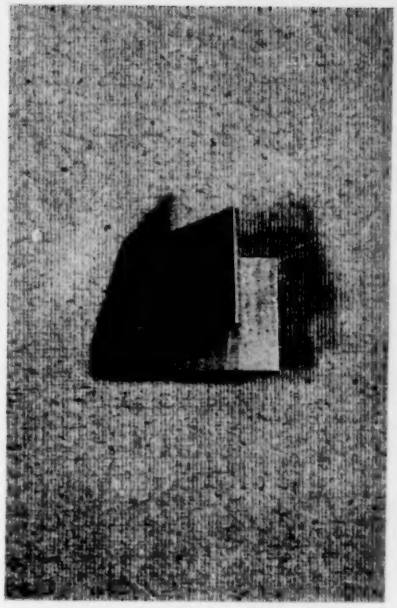

FIG. 2.

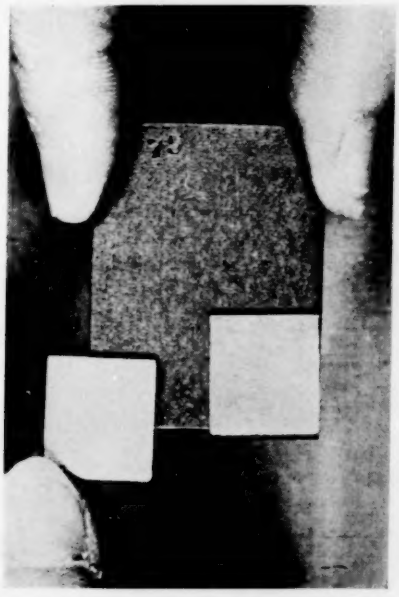

FIG. 3.

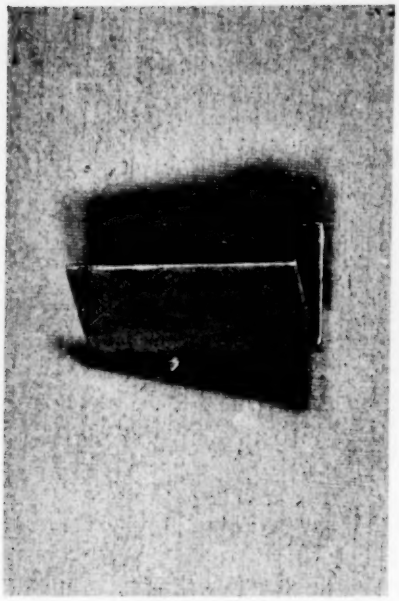

FIG. 4 (a).

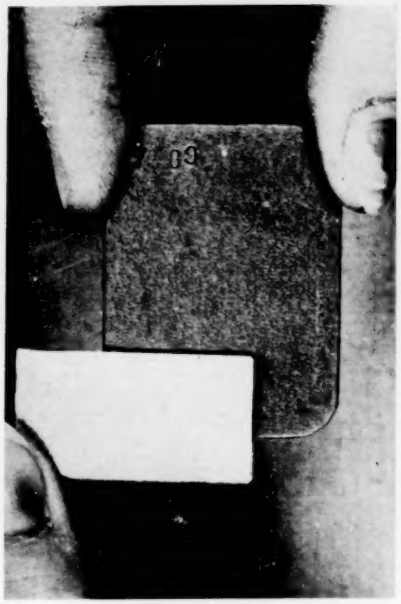

FIG. 4 (b).

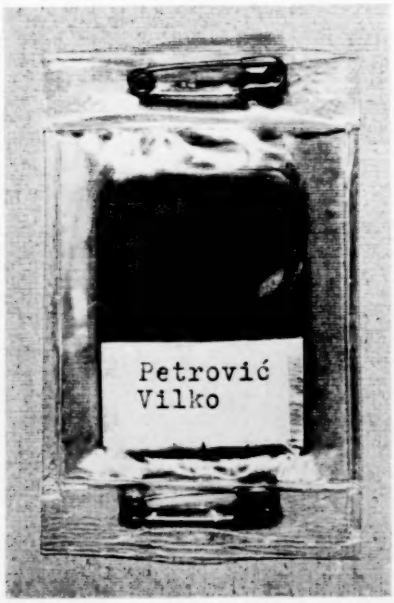

FIG. 5.

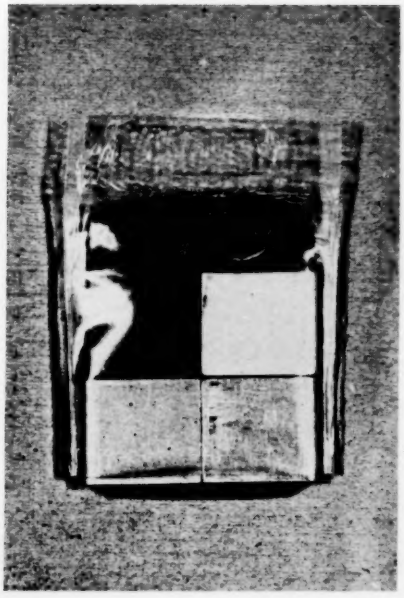

FIG. 6.

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NEWS

News Editor: J. J. FITZGERALD

AEC revises federal regulations on radiation exposure

The Atomic Energy Commission has revised its regulations for the protection of employees in atomic energy industries and the general public against hazards arising out of the possession or use of AEC-licensed radioactive materials. The revisions are embodied in amendments to Title 10, Chapter 1, Part 20, of the Code of Federal Regulations entitled "Standards for Protection Against Radiation." The amendments become effective on January 1, 1961.

Proposed amendments, designed to bring the Commission's regulations into accord with the current recommendations of the National Committee on Radiation Protection and Measurements, were published by the AEC in the Federal Register on May 2, 1959, for public comment.

These amendments refer specifically to licensed users of sources of ionizing radiation. AEC contractors have been operating essentially under the conditions of the principal amendments for more than two years.

The principal effect of the amendments will be to limit the life-time accumulated dose of workers in radiation areas to approximately one-third the limits permitted under the regulation as it now stands. The amendments will limit the total external radiation exposure that any worker may accumulate beyond the age of 18 to an average of 5 rems/year and to not more than 3 rems in any one quarter.

The amendments provide a basic table showing quarterly levels of radiation to which all workers may be exposed—$1\frac{1}{4}$ rems per calendar quarter for whole body, head and trunk, active blood-forming organs, lens of eyes or gonads; $18\frac{3}{4}$ rems for hands, forearms, feet or ankles; $7\frac{1}{2}$ rems for the skin.

The license may permit an employee to receive a greater dose than that listed in the basic table, provided that the quarterly dose to the whole body from radioactive material and other sources of radiation in the employer's possession does not exceed 3 rems and the dose to the whole body—when added to the employee's accumulated dose of radiation in previous jobs—does not exceed that arrived at by a formula based on the individual's age.

To permit exposure above the limits in the basic table, the employer first must make reasonable attempts to obtain reports of the employee's previous occupational exposure to radiation and must take these exposures into account in determining how much additional exposure an employee may receive.

In the absence of exposure records for an employee, the employer is to be guided by a table of assumed occupational exposure for the worker.

Notification and information supplied to workers

Employers will be required to notify the individual, as well as the Commission, of any exposure of the individual to radiation or to radioactive material above the established limits. Individuals who receive exposures in excess of quarterly limits (in the case of external radiation) or weekly limits (in the case of concentrations of radioactive materials) must be removed from further exposures during the remainder of the period involved.

The Commission may also require that, where appropriate, the employer provide an employee with an examination to determine the extent of his exposure to radioactive material. The employer must make the results of the examination known to the Commission.

Employers are required periodically to advise employees of their exposures if such information is requested by them. Forms are incorporated in the regulation for the recording of occupational exposures and for the recording of histories of exposure.

Employers must also post in conspicuous places an informative poster, which will advise employees as to the nature and purpose of the Commission's regulatory requirements, how they may get information as to these requirements, the reports which must be furnished to employees by employers, and similar matters.

Each licensee is also required to post or to make available for examination by employees upon request a copy of the Commission's Part 20 regulations and a copy of his AEC license, including any operating procedures applicable to the work permitted under the license.

Maximum permissible concentrations revised

The amendments include a comprehensive revision of the concentrations of radioactive material to which

employers may expose persons in areas under their control or which may be released by the employer into the environment without specific approval by the Commission.

In reference to persons in controlled or restricted areas and with respect to most isotopes listed, the principal changes in the values set forth in the new tables are a reduction to one-third in the concentrations of those radioisotopes having their principal effect upon the gonads or the whole body and the lowering of others to control exposure of the gastrointestinal tract to 15 rems/year.

The Commission's regulations are designed to make it unlikely that individuals in unrestricted areas receive exposure in excess of 10 per cent of the limits established for radiation workers.

All AEC-licensed activities are subject to inspection by the Commission.

AEC proposed revision of regulation controlling source material

The Atomic Energy Commission has published for public comment a proposed revision of its regulation governing control of source material (thorium and natural and depleted uranium), Title 10, Code of Federal Regulations, Part 40.

Under the revised regulation, possession of up to 15 lb of source material at any one time by a physician, pharmacist or research institution, or by persons receiving such material in medicines or drugs, would be generally licensed. The same 15 lb general license would apply to commercial and industrial firms or medical institutions using source material for educational, developmental or commercial purposes. Receipt of source material by a generally licensed person would be limited to 150 lb a year.

The proposed regulation would include a general license to receive title to source material. This would facilitate transactions in which the buyer does not take actual physical possession. There would also be a general license to export quantities of up to three pounds at any one time to specified destinations. Specific licenses would be required for export of larger quantities.

Certain finished products containing uranium or thorium would be exempt from domestic licensing requirements. These would include incandescent gas mantles, vacuum tubes, and welding rods, as well as some types of glazed ceramic tableware containing not more than 10 per cent of source material. Photographic films, negatives and prints containing uranium or thorium would be exempt; so would any finished product made of or containing tungsten–thorium alloys not exceeding 4 per cent thorium by weight, provided no treatment or processing was involved. The recently announced exemption from licensing requirements with respect to uranium contained in aircraft counterweights would be continued.

Very low radiation exposure at Hanford

Radiation exposure records for GE employees at HAPO show not one worker in a radiation area has received as much as one-third of the maximum permissible accumulated dose of penetrating radiation.

Analysis of the records shows that more than two-thirds of the 7803 people employed by GE at Hanford at the end of 1959 had accumulated external penetrating radiation doses totaling less than 1 per cent of the maximum permissible dose recommended by the NCRP. More than 90 per cent of the Hanford workers had accumulated radiation doses amounting to less than 10 per cent of the recommended maximum permissible dose.

GE reports advances in waste management and contamination control

GE-APED, San Jose, California, has designed and installed a special incinerator to dispose of paper clothing which will be used in the fuel shop. Disposal of said contaminated waste normally costs about $5.50 per cubic foot, according to GE. The new method of disposal is expected to lower this cost by 75 per cent. Presumably any combustible material may be disposed of in the incinerator which is designed to burn 30 cubic feet of material per hour (100 lb).

Disposable paper suits made of heavy-duty, fire- and water-resistant nylon threaded paper, are being worn in controlled areas of GE's Nuclear Fuel Manufacturing Operation as well as in the Vallecitos Atomic Laboratory. The paper suits cost about $1.00 each. This is considerably less expensive than having normal work clothing cleaned in a special laundry. The suits are worn once or twice, then burned in the incinerator described above.

Safe levels of ruthenium-106 on the loose

Some ruthenium metal distributed commercially in the United States has been found to be contaminated with small amounts of Ru-106. Calculations and measurements indicate that the contaminated material so far located does not present a health problem. Some of the material was introduced in dilute form in an alloy used in the manufacture of metal inserts for a type of artificial tooth marketed in the last few weeks. The teeth have been taken off the market.

Some of the material was used in dilute form in alloys by jewelry fabricators. The AEC is investigating

to determine the extent, if any, to which the material has been used in products distributed to consumers.

Zn-65 *detected in cyclotron workers*

Small but persistent amounts of Zn^{65} have been reported in cyclotron workers. VAN DILLA and ENGELKE at Los Alamos Scientific Laboratory reported in *Science*, March 1960, that amounts less than 1 per cent of the maximum permissible body burden for Zn^{65} can be detected using a human body counter. Amounts as small as 0.0035 μc were reported. The data indicated an effective half life approximately equal to the physical half life. In essence, the biological half life was considerably longer than the value contained in the ICRP Handbook. A request was made in this report for other installations to report their findings in cyclotron areas from the viewpoint of arriving at a better understanding of the metabolism of Zn^{65} in the human body.

AEC health physics courses for representatives of state governments

The Atomic Energy Commission will offer intensive courses in health physics for representatives of state and local governments as a step in encouraging the states to assume control of certain radioactive materials under an amendment to the Atomic Energy Act, passed in September, 1959.

This instruction will provide work experience for persons who will be concerned with licensing and inspection functions in their states.

The Commission's Health and Safety Laboratory in New York began a 10-week course in health physics on October 17, 1960, and will repeat the course in February of 1961. Argonne National Laboratory, Argonne, Illinois, expects to offer the course early in 1961.

Similar instruction will be provided at Oak Ridge, Tennessee. Oak Ridge Institute of Nuclear Studies will offer a four-week course on radioisotope techniques starting January 9, 1961, and again on March 6, 1961. This will be followed by a six-week course in health physics at Oak Ridge National Laboratory beginning April 4, 1961.

The AEC plans to make these health physics courses available in other parts of the country at a later date.

The Commission provides two other types of instruction for state personnel. They are:

(1) A one-year course in radiation control consisting of an academic year at the University of Michigan or Harvard University followed by three months on-the-job training in AEC facilities. The Commission pays tuition and all laboratory fees for students accepted for this course. Applicants should have a bachelor's degree with adequate preparation in science, preferably with mathematics, through calculus. However, majors in public health who may not fully meet these academic criteria will be considered.

(2) On-the-job orientation and work experience in the AEC licensing and regulation programs also are offered by the Commission to representatives of states planning assumption of the regulatory responsibility for the specified radioactive materials. This consists of two areas of instruction. The first is a period of two to three weeks of orientation of key administrative personnel of state regulatory authorities with AEC licensing, inspection and compliance functions, both at AEC Headquarters at Germantown, Maryland, and at an AEC Operations Office. The second area is on-the-job work experience for technically qualified personnel for four to eight weeks in licensing functions at AEC Headquarters and in inspection functions at an AEC Operations Office.

Survey of public attitude on atomic energy made by GE

The findings of the General Electric Company nation-wide public opinion survey on atomic energy are summarized.

(1) Acceptance of atomic power installations is generally higher in the West among men. The most negative reactions come from women in the South.

(2) Approximately 40 per cent have strong opinions on atomic energy (either negative or positive) and 60 per cent lack strong opinions and information on the subject.

(3) The average American is relatively un-informed and confused about the application of atomic energy for peaceful uses. The large majority preferred to receive information on the subject in "home town newspapers" and magazines. Less than 25 per cent preferred to be informed through books, television, radio and pamphlets.

(4) Approximately 25 per cent of the persons interviewed expressed positive comments in the hope that atomic energy would be developed to the fullest capacity for the public's benefit. Approximately 12.5 per cent had a desire for more knowledge. Ten per cent were opposed to the development of nuclear power.

NBS plans high-voltage accelerator studies

A 100-MeV linear electron accelerator is being built by High Voltage Engineering Corporation for

the National Bureau of Standards at the NBS Gaithersburg, Maryland, laboratories.

The 100-ft accelerator will produce an electron beam with peak energies of up to 150 MeV and power outputs of 40 kW or more. Segments of the device will be tested by Applied Radiation Corporation, an affiliate of High Voltage, before installation at the NBS Radiation Physics Laboratory late in 1962. It will be used for primary research in nuclear physics and in the development of dosimetry standards and measurement techniques.

Parker cites need to reduce medical dose

In August at Brussels, HERBERT PARKER of General Electric Company's Hanford Laboratories described as "striking" the high proportion of total dose to the whole U.S. population from medical exposure. While recognizing the desirability of preserving professional freedom in the medical uses of radiation, he suggested that "a voluntary reduction by 10 per cent, which must be easily achieveable, would compensate for all other forms of man-made radiation".

He added that if the radiation dose to reproductive organs "in medical procedures only were measured, practically all the man-made increment in the nation would be recorded. Such a step would serve to draw constant attention to opportunities to reduce that dose".

MIT radiation effects on radiosensitive foods announced

The results of a study by SAMUEL A. GOLDBLITH of the Massachusetts Institute of Technology on the effects of radiation from strontium-90 and cesium-137 on orange juice, selected as a typical radiosensitive food material, and on vitamin *C*, selected as a radiosensitive compound, have been announced by the AEC.

GOLDBLITH said: "Using the most sensitive techniques available and with extreme care and diligence in the numbers of samples exposed, tests employed, and techniques used, the data in this report indicate no public health hazard insofar as food products are concerned in the use of ionizing energy for process control."

In reference to this study and the reported results, the Food and Drug Administration stated that: "From a study of the data you have submitted, we conclude that use of devices of the activity employed by you (20 mc of Sr^{90} with a dose rate of 1·5 r/min and 150 mc of Cs^{137} with a dose rate of 632 mr/min) will not: (1) induce radiation in the exposed food products, or (2) destroy or diminish food constituents, when they are employed for the type of food-processing controls which we have had an opportunity to consider." Added the FDA: "While we are satisfied with the data on the safety of these devices for use in food processing controls under the limitations given above, these data do not in themselves establish the legality of the devices in actual manufacturing processes."

Health physics activities in Italy—ENRICO VIA

Need for radiation protection personnel. At the High Institute in Rome, Professor B. W. WINDEYIER, Director of the Meyerstein Institute at the Middlesex Hospital in London, spoke on the "Medical Aspects of the Nuclear Epoch". Professor WINDEYIER cited a need for: a small group of highly specialized experts that are capable of making high-level decisions; a larger group of less qualified (B.A. or B.S. level) to assume more routine assignments; and a still larger group of specialized technicians to operate under the intermediate group. The realization of a plan of this type would require the recruitment of personnel that are capable of assuming these responsibilities.

Insurance against radiation injuries. In Italy, as far back as February, 1958, a law was passed that involved compulsory insurance for physicians with respect to injuries caused by X-rays and radioactive substances. The insurance covers all medical practitioners and not only radiologists. However, the rules and regulations enabling the enforcement of the law have not yet been published.

Italian X-ray technicians formed a trade union and held, in 1959, a National Convention requesting that the directors of radiological laboratories insure their personnel against professional risks.

Health physics activities in France—F. DUHAMEL

European nuclear forum established. The national industrial nuclear associations of France, Belgium, Luxembourg, Italy, Germany and the Netherlands have agreed to co-ordinate their activities and to establish a Forum Atomique Européenne or in short—FORATOM. The association in Switzerland has made an immediate application for membership and has been unanimously accepted, and equivalent organizations in Denmark and other European countries have also expressed interest.

International food irradiation information centre. An international information centre for food irradiation has been established at Saclay in France following an agreement signed between the OEEC European Productivity Agency and the CEA. Functions of the centre include the preparation and publication, in English and French, of a quarterly international newsletter on food irradiation. The centre will serve

as a point of contact between scientists, food technologists and governmental and industrial circles. The contract covers a two-year period and provides for the payment of NF32,000 towards expenses.

Final agreement of Ispra granted. Final ratification by the Italian Government of the agreement concluded with Euratom for the transfer of the Ispra Nuclear Research Centre to Euratom has now been given. Euratom research teams have moved from Brussels to Ispra on Lake Maggiore, 35 miles from Milan. They are the vanguard of a total staff of 1500 research workers expected to be in the employ of Euratom by the end of 1962. More than 40 million dollars is to be spent in enlarging the centre over the next two years. Euratom Joint research centres are also being established at Petten in the Netherlands, at the Transuranic Elements Institute at Karlsruhe, West Germany, and at the Central Nuclear Measurements Bureau at Mol in Belgium. A European school along the lines of the ones at Luxembourg and Brussels will also be set up by Euratom at Ispra.

Health physics activities in Norway—LEIV BERTEIG

Nationwide film monitoring in Norway (Statens radiologisk-fysiske laboratorium, Oslo, Norway). The efficiency of a national radiation protection service can, at least partially, be established by measurement of all professional radiation doses in a nation. This means monitoring of the radiation load to medical, dental and veterinary personnel, to scientists and students in radiation research, to certain groups in industries employing ionizing radiation and to persons occupied in atomic energy installations.

Table 1.

	mrems/year
Radiologists	381
Radiologists' assistants	377
Surgeons	52
Surgeons' assistants	432
Lung specialists	121
Lung specialists' assistants	159
Radium personnel	1985
Dentists and assistants in private clinics	173
Dentists and assistants in public clinics	2350
Veterinarians and assistants	412
General industry	107
Industrial gammagraphy service	1920
Reactor personnel*	580

* Data obtained from Institutt for Atomenergi.

The results of such a nationwide film monitoring system can be presented in several ways. We have tried to give a picture of the present professional radiation exposure situation in Norway by grouping the radiation personnel into what we think representative categories and stating their relevant mean yearly radiation doses. Our data for 1959 are as in Table 1.

We think that our observations show that it should be relatively easy to comply with the latest maximum permissible doses recommended by the International Commission on Radiological Protection for those who are professionally exposed to external radiations. It is our experience that the necessary measures for improved radiation protection do not lead to excessive expenditures in installations.

There is, of course, still room for certain improvements, and we expect that our data for next year will be even better. They certainly will for dentists in public service, and probably also for those in private clinics.

Gonad doses. In 1959, the State Institute of Radiation Hygiene (Statens radiologiskfysiske laboratorium) in Norway carried out a series of gonad dose measurements intended to give an estimate of the genetically significant gonad dose to the population due to medical X-ray diagnostics. Measurements at eleven different X-ray departments and some smaller establishments were performed. The method applied was to combine rectal measurements in patients with phantom measurements, and approximately 1000 single measurements were done. Data on the number of examinations distributed according to examination type, age and sex were taken from hospital journals and from our questionnaires, returned to us from most X-ray installations in Norway.

The results of our investigations show that approximately 2,200,000 X-ray examinations were performed in Norway in 1958, the given number not including dental X-ray examinations. The final calculation of the genetically significant dose from X-ray diagnostics has not yet been fully completed, but a provisional calculation leads to a value of 22.8 mr for 1958. The accuracy of this result is not too high, probably of the order of 25–35 per cent.

Neutron dose determination—Na^{24} *in blood.* As a part of the plans for actions in case of overexposure to personnel, particularly in case of neutron exposure, a technique for the determination of sodium-24 in blood has been investigated. Some already existing liquid scintillation well counters have been calibrated and showed that a sodium-24 content in blood of 2 $\mu\mu$c/ml can easily be detected within a few minutes counting directly on a 5-ml blood sample.

As the doses received depend largely on the actual

neutron spectrum, the use of threshold detectors and other monitors is also being studied.

Medical treatment of possible patients will take place at the Rikshospital, Oslo.

Health physics activities in Japan—Toshio Aoki

The first Japanese Health Physics Organization was established at the Japan Atomic Energy Research Institute (JAERI) in June 1956, one year prior to the operation of the first nuclear reactor in Japan. Health Physics activity has since grown very rapidly in Japan. At present this is the largest and most active health physics organization employing about sixty-five personnel, although in the beginning there were only eight health physicists. The Health Physics Division is divided into two sections, radiation control and health physics research. The research and engineering programs are radiation dosimetry, internal dosimetry, bio-assay analytical technique, decontamination technique, low-level counting, total body counter, shielding, development of instruments and meteorological and oceanographic studies. The Radiation Safety Manual is now being revised to comply with the 1959 recommendations of the ICRP, and will also include an emergency plan for radiation accidents.

First certification of health physicists effective December 1960

A statement by Mr. W. McAdams, Chairman of the American Board of Health Physics, indicates that approximately 140 certifications will be awarded to health physicists in the early part of December 1960. The American Board has processed more than 200 additional applications. These applications, however, have not been fully appraised at this time.

The approximate 140 certificates will be awarded to those health physicists who were approved initially without the need for an examination and to those applicants who have successfully passed the first examination that was taken in Boston prior to the Health Physics Meeting. The second examination will be given in December 1960.

Certification has been delayed because of the need for incorporation of the Board to avoid potential legal problems.

Approximately 400 applications have been received and are being processed as rapidly as possible.

Dosimetric investigation of the Yugoslav radiation accident (*From* G. S. Hurst—*ORNL*)

A complete investigation of the radiation doses received by individuals exposed to radiation in the Yugoslav accident[1] on 15 October 1958 has been completed, using modern techniques of dosimetry. The zero power reactor[2] was operated at low power levels to determine the ratio of γ-ray to neutron dose at various locations near the reactor. These measurements were made with special counting methods, one of which measured the neutron dose in the presence of γ-rays,[3] and the other measured the γ-dose[4] in the presence of neutrons. The reactor was operated at a higher power level for a short time to determine (a) the neutron dose by means of threshold detectors[5] and (b) the relationship between neutron dose and Na^{24} activation in man-shaped phantoms.[6] A computational program gave theoretical results for the same three quantities, i.e. the neutron spectrum, the gamma–neutron ratio, and Na^{24} activation. The results of theory and experiment are in good agreement.

Final results for the individual doses are shown in Table 1. In this table the charged particle dose refers to the first collision dose due to recoils from fast neutrons, and includes the dose delivered by the nitrogen n.p. reaction. The second dose column gives the maximum dose obtained for the hydrogen capture γ-radiation produced in the phantoms. The external γ-dose based on the measured gamma–neutron ratio is shown in the third column, and the total doses are given in the last column. Of the seven persons investigated, the highest dose received (individual *V*) was 436 rads and the lowest dose was 207 rads. The doses previously estimated were reported in rem units, showing for individual *V* an estimated value of 210 rems for the neutron dose and 630 rems for the γ-dose, or a total exposure of 840 rems.

Table 1. *Individual doses* (*all values are in* rad *units*)

Individual	Charged particle dose	H(n,γ)D γ-dose	External γ-dose	Total
H	66	99	158	323
V	89	133	214	436
G	90	135	189	414
M	87	130	209	426
D	91	136	192	419
B	45	67	95	207

A complete report of the dosimetry experiment will be given in Vol. 5, No. 1 of *Health Physics*, and is also available through the International Atomic Energy Agency, Vienna 1, Kaerntnerring, Austria.

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FIFTH ANNUAL MEETING OF THE HEALTH PHYSICS SOCIETY

29 June to 1 July 1960 at Boston, Massachusetts

Abstracts and titles of papers presented at the meeting

Symposium on Sea Disposal. Chairman: E. G. STRUXNESS, ORNL, Oak Ridge, Tenn.

Disposal of radioactive wastes in the ocean. DONALD W. PRITCHARD, Johns Hopkins University, Baltimore, Maryland.

Fundamentals of dispersion and diffusion in the ocean. ROBERT S. ARTHUR, Scripps Institution of Oceanography, LaJolla, California.

Fundamentals of ecology in the marine environment. MILNER B. SCHAEFER, Inter-American Tropical Tuna Commission, LaJolla, California.

Legal and administrative aspects of ocean disposal. ROBERT LOWENSTEIN, Office of General Council, U.S. AEC, Washington, D.C.

Contributed Papers on Dosimetry. Chairman: W. T HAM, JR , Medical College of Virginia, Richmond, Va.

Dosimetry in a reconstruction of the reactor excursion in Yugoslavia. J. A. AUXIER, Health Physics Division, ORNL, Oak Ridge, Tenn.* (Invited paper)

An accidental excursion of the zero power reactor at the Boris Kidric Institute near Belgrade, Yugoslavia, occurred on 15 October, 1958. Six persons received doses of radiation estimated at the time to be in the mid-lethal to supralethal range; one of the six died about 3 weeks after the accident. During April 1960 a dosimetry team from the Health Physics Division of the Oak Ridge National Laboratory conducted experiments at the Boris Kidric Institute to determine the exposure doses of these six people. A team of French nuclear engineers from Saclay modified the reactor for safe operation at power levels up to 5 kW, and a scientist from the Atomic Energy Research Establishment at Harwell, England, joined the team in Yugoslavia. The staff of the Boris Kidric Institute collaborated fully with the other teams. The International Atomic Energy Agency co-ordinated the various phases of the experiment. The experiment included the determination of the neutron to γ-ray ratios (first collision doses in rads) at points throughout the area, the neutron flux in various energy increments, and the activation of sodium in man-sized phantoms filled with a homogeneous solution of NaCl. The phantoms were placed at positions at which the exposed persons were estimated to have received the greatest portion of their doses, and the activation of the sodium was related to the first collision neutron dose at that point. Other detectors were used to determine the neutron to γ-ray ratios at the same points. After the accident the percentage of activated sodium in the blood had been determined for each of the individuals at Saclay. Consequently, with the information obtained in these experiments, the fast neutron dose could be calculated from sodium activation and the γ-dose from the neutron to γ-ray ratios at the points of interest.

An emergency criticality dosimeter. J. E. HOY, Savannah River Plant, E. I. du Pont de Nemours and Company, Aiken, S.Ca.

A small, low-cost criticality monitoring device was designed to measure neutron and γ-exposures which might be received by personnel involved in a nuclear incident or an accidental excursion. Test exposures indicate that the device is suitable for measuring neutron exposures above 1 rad and γ-exposures between 50 r and 10^4 r.

Angular response of a partially shielded cylindrical dosimeter. R. D. BIRKHOFF and WILLIAM HAPPER,* Health Physics Division, Oak Ridge National Laboratory†, Oak Ridge, Tennessee.

Some cylindrical dosimeters have very thin walls and are partially shielded by a reinforcing structure. If the tops and sides of the dosimeter are unequally shielded the response of the dosimeter will depend upon the angle θ between the plane of the cylinder base and the direction of the incident radiation. For instance, a cylinder with unshielded top and opaque sides will admit all radiation

* Operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

* Present address: University of North Carolina, Chapel Hill, N.C.
† Operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

incident perpendicular to the plane of the base, but none which is incident parallel to the base. The fraction of volume irradiated through the top and the fraction irradiated through the sides are calculated separately in analytical form as functions of the angle θ and the ratio of cylinder height to radius, h/p. The response of the dosimeter may then be found from these functions and the percentages of the top and side areas which are shielded.

Response of anthracene as a function of LET and its use in a LET meter. R. D. BIRKHOFF and R. M. JOHNSON, Health Physics Division, Oak Ridge National Laboratory,* Oak Ridge, Tennessee.

The specific fluorescence dS/dx of anthracene has been shown by BIRKS to depend on the specific energy loss dE/dx for protons, α-particles, and high-energy electrons as follows:

$$\frac{dS}{dx} = \frac{A(dE/dx)}{[1 + B(dE/dx)]}$$

where the constants A and B are the same for all three radiations. Evaluation of data taken at this Laboratory on the pulse height-energy characteristic for monoenergetic low-energy electrons (10 keV → 125 keV) shows that this data also is in accord with the above equation. The generality of this relationship has led us to consider the application of the non-linear response of anthracene to the development of a LET meter. Such a device may be constructed from a sandwich of a thin layer of anthracene and a thin inorganic phosphor or proportional counter. A particle traversing these layers at any angle will cause a pulse height proportional to dS/dx in the anthracene and proportional to dE/dx in the other layer. The dE/dx of the particle may then be computed from the ratio of pulse heights (which is independent of the angle of traversal) and the above equation. Experiments have been conducted with an anthracene layer 0.2 mg/cm² thick evaporated in a vacuum on to the face of a 6199 photomultiplier. Pulses from α-particles are obtained which are readily distinguishable from the background and indicate that such a device may be feasible.

A polyacrylamide γ-dosimeter. A. L. BONI, Savannah River Plant, E. I. du Pont de Nemours and Company, Aiken, South Carolina.

A simple chemical dosimeter system which will measure only the γ- and X-radiation in the dose range between 50 r and 7500 r has been developed and evaluated for both criticality and general radiation monitoring. The dosimeter is based on the degradation by ionizing radiation of a high molecular weight homopolymer of acrylamide in an aqueous solution, and is unaffected by fast and thermal neutron fields. Dosage is determined from the change in viscosity of the polymer solution. The stability of the polymer both before and after irradiation enables it to accumulate dosage from intermittent exposures.

* Operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

A G–M tube γ-ray dosimeter with low neutron sensitivity. E. B. WAGNER and G. S. HURST, Health Physics Division, Oak Ridge National Laboratory*, Oak Ridge, Tennessee.

It has been shown that a commercially available halogen type Geiger–Müeller counter when properly shielded with Sn and Pb provides a very practical and satisfactory device for measuring the dose due to γ-radiation in the presence of neutrons. The device measures the dose in roentgens with nearly uniform response for effective X-ray energies from 200 keV and up to at least 2.0 MeV γ-energies. Calculations show that the fast neutron response is less than 0.15 per cent. Experiments show that about 5×10^9 thermal neutrons per cm² give a response equivalent to 1 r of γ-radiation. When necessary, the thermal neutron response can be decreased by a factor of 300 by using a Li shield.

Some studies of film dosimeter variables. W. V. BAUMGARTNER, Hanford Laboratories, General Electric Company, Richland, Washington.

Latent image fading was measured for duPont 552 film and Kodak Personnel Monitoring Type 2 film for storage periods from 1 day to 1 year. Film developing techniques were analyzed and estimates of the variables were made. A statistical approach to determine the radiation detection levels for film is presented.

Conductivity changes in insulating materials as a mechanism of radiation dosimetry.† D. C. LAWRENCE, Atomics International, Canoga Park, California.

An investigation into the conductivity properties of polyethylene in a radiation field is reported. It is shown that the radiation-induced conductivity increases as the 0.80 power of the exposure dose rate from 30 mr/hr to 10^4 r/hr. Application is indicated for use as a solid-state mechanism of radiation detection. A "memory" effect in the radiation-induced conductivity is also described. Implications in the design of insulators for radiation detectors are discussed.

Energy response and physical properties of NTA personnel neutron dosimeter nuclear track film. R. L. LEHMAN, Lawrence Radiation Laboratory, Berkeley, California.

We report for NTA personnel neutron dosimeter film: the dose-response relation for thermal, PoBe and PuBe neutrons; the response of neutrons of specific energies 0.2, 0.4, 0.6, 0.8, 2.4 and 14.5 MeV; and the variation in response with isotropic, end-on and normal exposure to PuBe neutrons. In addition, we report a chemical analysis of NTA emulsion, and the correlation between thermal neutron response and nitrogen density.

We present a part-empirical, part-theoretical energy vs.

* Operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

† Based on part of the work done for a master's thesis in Nuclear Engineering at the University of Washington, 1958.

response curve for NTA dosimeter film exposed to neutrons between thermal and 100 MeV energies. In addition, we present an analysis of the proton-radiating and proton-stopping power of the various layers in the NTA film packet, and we present a brief general discussion of the interaction of neutrons with NTA nuclear track emulsion.

Contributed papers on Air Cleaning and Atmospheric Contamination. Chairman: H. F. SCHULTE, Los Alamos Scientific Laboratory. Co-Chairman: L. SILVERMAN, Harvard University.

Symposium on the "Effect of the NCRP Recommendations on National Life." Chairman: H. P. YOCKEY, Aerojet-General, San Ramon, California.

Effects on labor. C. FREDERICK MACGOWAN, International Representative, AFL-CIO, New Brotherhood Building, Kansas City, Kansas.

Legislative aspects of radiation standards—Role of NCRP and Government. Congressman CHET HOLIFIELD, Washington, D.C.

Effects on medicine. Dr. SHIELDS WARREN, Cancer Research Institute, New England Deaconness Hospital, Boston, Massachusetts.

Effects on insurance. CHARLES R. WILLIAMS, Liberty Mutual Insurance Co., Boston, Massachusetts.

Problems in nuclear aerospace applications. D. M. ROSS, U.S. Atomic Energy Commission, Washington, D.C.

The use of nuclear energy in space raises the possibility of introducing new sources of radiation to the earth's atmosphere and oceans, to other solar system bodies, and into areas along probable orbit paths. It is felt that the AEC must insure that the hazards of devices developed under its cognizance be minimized to the greatest practical extent and that these devices can be, and are, safely used.

In order better to define the problems and make a proper start toward their solution, an "Aerospace Nuclear Safety Board" was established to:

(1) Analyze and project the effect of nuclear space devices upon the health of the peoples of the world.

(2) Analyze (and maintain cognizance over) the present and future radiations from space nuclear power sources on earth and in those outer space regions of special interest which are within the prerogatives of the United States.

(3) Recommend standards of safe practice for the employment of nuclear powered space devices proposed by the United States.

(4) Recommend policy and procedures to be followed in regulating the use of nuclear energy for devices in space vehicles and satellites proposed for use by the United States.

The hazards associated with the following programs are being studied by the Board:

(1) Rover, pointed toward demonstrating the feasibility of applying nuclear energy to rocket propulsion.

(2) Pluto, a study of the feasibility of a nuclear reactor to power a ramjet.

(3) SNAP, the development of nuclear energy into light-weight, long-life electrical power units for space applications.

(4) Manned nuclear aircraft.

What kinds of problems must be solved before nuclear energy can be introduced into space? Here are samples:

In the case of radioisotopic heat sources destined for orbital use on satellites is the package designed to withstand every conceivable launch abort and remain intact?

With what assurity can it be said that a satellite which fails to reach orbital velocity will *not* drop intact on a land mass?

Until the state of the art permits the controlled re-entry of satellites, can it be guaranteed that the isotope package or reactor will burn up upon orbital decay re-entry into the earth's atmosphere?

If the isotope package is burned up upon re-entry, what is the nature of the resulting particulates and can it be assumed that they will be disseminated as bomb fallout?

What are the parameters which dictate the selection of a site for testing nuclear powered ramjets or manned aircraft?

Recent developments in meteorology as applied to atmospheric pollution. MORTON L. BARAD, Cambridge Research Institute, Cambridge, Mass.

The results of studies in atmospheric dispersion in non-uniform wind fields and terrain will be discussed briefly. In addition, there will be a more detailed discussion of dispersion in uniform wind fields and terrain. In the latter discussion, the patterns of both time-mean concentrations and short-period concentrations downwind of both ground and elevated sources will be summarized.

Particle size studies on plutonium aerosols. W. D. MOSS, E. C. HYATT and H. F. SCHULTE, Los Alamos Scientific Laboratory, Los Alamos, N. Mex.

Particle size is recognized as an important factor in evaluating the health hazard of aerosols and, in addition, has other applications in health physics. However, very little published data is available on plutonium aerosols produced in plant and laboratory operations. Since the concentration of plutonium in most such aerosols is extremely small relative to the normal dust loading of the atmosphere, direct identification and measurement of plutonium particles is impossible. In these studies an α-track technique has been used. The sample is collected on a membrane filter and transferred directly to a nuclear emulsion where it is allowed to remain in contact for various time periods. After development of the film, the number of tracks per particle are counted with the microscope. Assuming the particles are spheres of pure plutonium oxide, the particle diameter can be calculated from

the number of tracks. Limitations of the technique are discussed.

General air samples and breathing zone samples were collected in a processing plant where plutonium metal is handled in glove boxes. The operations conducted in the rooms included metallurgy and machining of plutonium. Particle sizes measured were mostly less than 0.3 μ and the standard deviations of the size distributions were less than 2.0. Since most samples were collected during normal plant operations, the small sizes found were not unexpected and were rather uniform despite the variety of operations covered. These data should have application in setting maximum permissible concentrations and indicate the need for further biological studies with very small plutonium particulates.

The deposition of particles in lines transporting aerosol streams with reference to aerosol sampling accuracy. A. K. Postma and L. C. Schwendiman, General Electric Co., Hanford Laboratories, Richland, Washington.

Accurate sampling of particulates in gaseous effluents is required to audit and control radioactive wastes released to the environment at installations generating or processing radioactive material. Due to the inaccessibility of some sampling points, sample delivery lines are frequently required to transport the sample to a collector at an accessible location. A knowledge of the magnitude of particle retention in the sample line is required if the validity of the sample is to be evaluated. The movement of particles toward the wall of the sample line due to thermal and gravitational forces, Brownian diffusion, and turbulent eddies in the flowing fluid is reviewed. Estimates are made of resulting particle deposition in tubes as a function of velocity, pipe diameter, and particle size and density. Theoretical considerations were used in the calculation for deposition resulting from Brownian diffusion, and gravitational forces. Experimental data are presented for turbulent deposition from studies performed by the authors and others. The "half-length" or length of delivery tube which will reduce the concentration by a factor of 2 is presented for a number of cases. It is concluded that depending on the nature of the material to be sampled, the sampling velocity and line configuration, significant errors in particle concentration measurements may occur.

Solids removal from waste calcination off gas. B. R. Wheeler and E. Hylsky, Phillips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho.

Development studies concerned with the clean up of off gas from the Waste Calcination Facilities currently being installed at the Idaho Chemical Processing Plant are described. Solids concentration in the off gas discharged to the stack must not exceed from 2 to 12 $\times$ 10^{-5} grains per ft^3 depending on weather conditions, because of the presence of fission products. These limits are based on allowable concentrations of strontium-90 at the perimeter fence as given by the NBS Handbook 69.

Calcination conditions determine the size and concentration of solids in the off gas. Tests with non-radioactive alumina indicate that the solids loading may be reduced to 0.5 $\times$ 10^{-5} grains using only a venturi scrubber and electrolytic precipitator in series if a pressure drop of 5 in. of mercury across the venturi scrubber can be tolerated. At this pressure drop the calciner venturi scrubber alone is capable of reducing solids loading to 160 $\times$ 10^{-5} grains per ft^3 of gas. Other data indicate that venturi scrubbers are capable of reducing solids loading to 7.5 $\times$ 10^{-5} grains per ft^3 at pressure drops of 18 in. of mercury.

The oxidation and release of iodine-131 from uranium slugs oxidizing in air and carbon dioxide. W. J. Megaw, R. C. Chadwick, A. C. Wells and Jennifer E. Bridges, United Kingdom Atomic Energy Authority, Harwell, Berkshire, England.

Experiments to determine the oxidation rate, particle size distribution of oxide and release of iodine-131 from 1 kg, trace irradiated uranium slugs oxidizing at temperatures up to 1000°C in air and carbon dioxide are described. The release of iodine as a result of rapid melting of uranium is also investigated. It is concluded that extensive release of elemental iodine as vapour is always accompanied by extensive oxidation and that for temperatures below the melting point the release of elemental iodine as vapour is greater for oxidation in air than in carbon dioxide. It is stressed that for the geometrical arrangement and gas velocities used the greater part of the iodine inventory of the uranium remains contained in or adsorbed on particles of uranium oxide.

Contributed papers on Health Physics Instruments. Chairman: J. B. H. Kuper, Brookhaven National Laboratory.

An improved dose integrating survey instrument. W. A. Higinbotham, E. Rainey and W. H. Bishop, Jr., Brookhaven National Laboratory, Upton, L.I., N.Y.

This instrument has been recently developed for measuring stray radiation around pulsed accelerators, in particular the BNL Cosmotron. It consists of a 3.75 l. ionization chamber and an integrating electrometer circuit. Full scale ranges are 7, 70 and 700 μrads which correspond to average dose rates of 5, 50 and 500 mrad/hr when produced by a single burst of radiation from the cosmotron with a repetition rate of 720 pulses/hr. The ionization chamber design utilizes a commercially available polyethylene jar and cover which make the instrument simple to construct. In addition, the chamber is much more nearly tissue equivalent than one constructed of Bakelite. The electrometer circuit is a novel one employing one electrometer tube and two transistors. Power for this circuit, as well as collecting potential for the ionization chamber, is supplied by a transistor d.c. inverter circuit. Two 1.3-V mercury cells supply both circuits for approximately 1000 hr of operation. A Victoreen VX-10 thermal switch is used to discharge the 10, 100 and 1000 pF input capacitors. Precision ($\pm$2 per cent) polystyrene foil

capacitors minimize leakage and allow calibration adjustment of three ranges with one potentiometer. The final packaging of the instrument features a plug-in circuit to facilitate maintenance.

A ten-channel radiation monitoring system for pulsed accelerators.* W. A. HIGINBOTHAM, E. RAINEY, J. CHESTER, Brookhaven National Laboratory, Upton, L.I., N.Y.

In anticipation of the radiation monitoring problems arising out of the expanded facilities and increased operating efficiency at the Brookhaven cosmotron, a ten-channel remote monitoring system has been designed and constructed. Each channel consists of a remote detector, analog-to-digital converter, recorder and alarm-type meter. The detector is a 5 l., air-filled polyethylene ionization chamber connected to an electrometer amplifier. The amplifier is an integrating type which is reset at the start of each cosmotron pulse. The signal is fed by cable from the remote unit to the analog-to-digital converter which is a gated oscillator producing a numerical value proportional to the input pulse. The scaler with a capacity of 10^9 counts adds the numerical value for each succeeding pulse, providing an integrated dose for any operating period. The height of each pulse, proportional to the dose, is recorded on a Rustrak recorder, facilitating a detailed study of the radiation pattern. A timing circuit synchronizes the operation of the monitor with cosmotron operation. Meter indication is provided at each remote unit and on the cosmotron console. The meters follow the output of the electrometer amplifiers returning to zero at the start of each pulse. The console meters are alarm type and the alarm levels may be set by the Health Physicist to meet operating limitations.

Equipment for self-monitoring in a plutonium facility. R. D. SMITH and V. P. JOHNSON, Health Physics Department, The Dow Chemical Company, Rocky Flats Plant, Denver, Colorado.

Early detection of contamination is one of the primary goals of a health physics group in a plutonium facility. While the responsibility for detecting personnel contamination falls primarily on the health physics group, a justifiable solution lies in compromise, that is, a program of self-monitoring concurrent with continual coverage by trained health physics personnel.

The effort to provide adequate self-monitoring equipment at Rocky Flats has resulted in three unique instruments, each suited for a distinct application. These instruments are: (1) a modification of the Eberline Instrument Company's PAC-1A; (2) Combo, a battery powered console type instrument; and (3) a modification of the NRD Instrument Corporation's CRM-560 count rate meter.

In routine usage, all three of the instruments are capable of showing the absence or presence of gross contamination.

The implementation of a self-monitoring program in conjunction with health physics monitoring personnel has proven to be quite successful.

* Work done under the auspices of the Atomic Energy Commission.

A continuous automatic air sampler for the determination of radon and thoron. GEORGE I. COATS, Robert A. Taft Sanitary Engineering Center, Cincinnati, Ohio.

The instrument described automatically determines the airborne α-activity. The radon and thoron daughter products are differentiated by decay counting. Long-lived α-activity can also be determined. The sampler consists of a membrane tape filter with an automatic advance transport. The filter tape can be advanced at predetermined time intervals from 0 to 24 hr. α-Activity is monitored by two ZnS photomultiplier detectors coupled to a duel channel ratemeter and strip-chart recorder and can be determined at the sampling head or any integral number of advance cycles later. The daughter product activities on the filter are mathematically related to the radon and thoron concentrations in the air. Effects of various operating parameters and preliminary data on environmental concentrations are given.

Stack and water effluent monitor systems. THOMAS L. JUNOD, NASA Lewis Research Center, Plum Brook Reactor Facility, Sandusky, Ohio.

The Water Effluent Monitor Station monitors all waste water prior to discharge to Plum Brook, to ensure compliance with Title 10 CFR Part 20. Either one of two proportioning pumps continuously samples a predetermined fraction of the total effluent flow. This fraction is then automatically monitored. If the activity exceeds a predetermined set point, sluice gates automatically close and an alarm is initiated. Also, if the Effluent Monitor System becomes inoperative, the sluice gates close. The monitored fraction of the total flow is stored, and a sample is analyzed every 8 hr by refined counting room techniques.

The stack effluent monitor continuously monitors the ventilation air from the containment vessel, reactor building and hot laboratory, and the waste cooling air and process off-gas systems, all of which comprise the stack effluent. Appropriate filter–detector combinations monitor particulate and radioiodine activity and other detectors monitor gross β–γ radiation and gaseous activity. Containment vessel air, prior to stack release, is discharged to monitored tanks. If the activity is within preset limits, the tank is automatically exhausted to the stack. If the activity exceeds this limit, an alarm is initiated and the tank contents are not released, uncontrolled, to the stack.

A proportional continuous river water and stream sampler at Georgia nuclear laboratory. J. R. WRIGHT, J. M. SELBY, B. M. BOWEN and J. H. EDGERTON, Radiological Safety Group, Lockheed Nuclear Production Branch, Dawsonville, Ga.

The main facility at the Georgia Nuclear Laboratory is an air shielded 10 MW reactor designed for a high leakage flux. This reactor located within 750 ft of the Etowah River is used for irradiation of aircraft systems and components. Because of the high neutron dose and the resultant soil activation, rigid control is maintained on run-off streams and the Etowah River. In order to overcome the lack of power at river and stream sampling sites, a

sampler was designed to utilize river power. The sampler, built on the water wheel principle, includes an adjustable suction device and a revolution counter. Using a current measuring apparatus such as a Curley meter and maintaining the sampler at a desired sampling rate, the total revolutions and the volume collected can be related to the rate of flow of the river. The construction of the continuous proportional water sampler and the calibration are discussed in this report.

Recent developments in OCDM radiological instruments. MARLOW STANGLER, Radiological Physicist, Office of Civil and Defense Mobilization, Battle Creek, Michigan.

A special reticle has been developed for low-range self-reading quartz fiber dosimeters which simplifies the determination of radiation dose rates. The new scale arrangement is incorporated in a dosimeter of nominal 2 r range. The readings are determined by the distance moved by the hairline indicator during a given period of time. To determine dose rates the instrument is first zeroed. After a 1 min timing interval, dose rates to 120 r/hr may be read directly on a scale. For more accurate determination of dose rate below 12 r/hr, a 10-min timing interval permits reading the dose rate directly on a second scale on the reticle.

A dosimeter equipped with the scale described is part of a three-instrument package recommended by OCDM for those who have constructed family fallout shelters. A companion quartz fiber dosimeter with a range of from 0 to 600 r, an associated charging unit, and a simple instruction manual complete the package of a total delivered cost of $20. The performance of these instruments is identical to other OCDM operational dosimeters with the exception of a relaxation in specified accuracy from ± 10 per cent to ± 20 per cent.

Radiological instruments of proven reliability, accuracy and simplicity are now available for public purchase. OCDM has development contracts directed toward further simplicity and cost reduction of radiological instruments. The family of standard radiation measuring instruments for operational use is being improved. These are discussed in further detail in the paper.

A **cobalt neutron flux integrator.*** ALAN R. SMITH, Lawrence Radiation Laboratory, Berkeley, California.

The fast neutron detector described here is expected to be most useful for health physics purposes. The detector is a cobalt disk embedded in a cadmium-cased paraffin moderator. The Co^{60} produced in a disk during irradiation is measured subsequently with a γ-ray scintillation spectrometer to determine the integrated neutron exposure. The present detector–spectrometer system can measure integrated exposures of 1×10^7 n/cm^2 or greater; the limit on minimum detectable flux integral is imposed by spectrometer background count rate.

* This work was done under the auspices of the U.S. Atomic Energy Commission.

The end-effect correction for measuring neutrons with stilbene scintillators. ALLEN BRODSKY, United States Atomic Energy Commission and GEORGE E. OWEN, Department of Physics, The Johns Hopkins University.

A detailed analysis has been made of the pulse-height distribution produced in a stilbene slab by recoil protons that escape the back surface, for incident neutron energies of 5, 10 and 15 MeV. The integrals of the calculated pulse distributions for a 0.5-cm-thick crystal, corrected for this "end effect" of escaping protons, agreed with the detection efficiencies calculated from cross-section data to within 1–2 per cent. The pulse distributions corrected for end effect were also compared with those calculated from the approximation of SWARTZ, OWEN and AMES:

$$n(P)\,dP = (\sigma_0\,DL/E_0)\left[1 + \frac{R_0}{L}\left(\frac{2}{5} - \frac{7}{5}\,(E/E_0)^{5/2}\right)\right](dE/dP)\,dP$$

where $n(P)$ is the pulse-height distribution in units of pulses/neutron per MeV-electron-equivalent pulse-height interval, P is the pulse height in terms of the MeV electron energy required to give the same pulse height, σ is the neutron–proton scattering cross-section at energy E_0 in units of cm^2/atom, D is the proton density in the scintillator in hydrogen atoms/cm^3, L is the length of the scintillator slab (cm), E_0 is the energy of the incident neutrons (monoenergetic) (MeV), R_0 is the range in stilbene of a proton of energy E_0 (cm), and E is the energy of the recoil proton that produces a pulse of size P when it expends its entire path in the scintillator. The above approximation was found to agree with the more exact calculations over the energy range 2–15 MeV to within 3 per cent for $L = 0.5$ cm and $E_0 = 15$ MeV, and to within 10 per cent for $L = R_0 = 0.2131$ cm and $E_0 = 15$ MeV. Thus, the above equation would be quite accurate for use in measuring neutron intensities, spectra and dosages with stilbene crystals. The calculated pulse distributions will be presented, and methods of deriving neutron intensities and spectra from experimental pulse-height distributions will be outlined.

The Federal Radiation Council.

The Federal Radiation Council.* Speaker: ARTHUR S. FLEMMING, Secretary of Health, Education and Welfare, Washington, D.C.

Contributed papers in Applied Health Physics Programs. Chairman: F. P. COWAN, Brookhaven National Laboratory.

A modernized film badge system. J. W. ADAMS and C. N. WRIGHT, Savannah River Plant, E. I. du Pont de Nemours and Company, Aiken, South Carolina.

An automated film badge system has been designed to provide a faster and more reliable means for recording radiation exposure received by personnel. Film packets are changed automatically and are X-ray marked for identification purposes. After manual development, films

* Talk presented by: DONALD R. CHADWICK. (Invited paper)

are automatically read, and then the badge number, the radiation dose and other data are punched on IBM cards.

In addition, a lighter and more durable film badge has been developed which aids in distinguishing between β- and soft γ-exposure.

Analysis of distribution of externally received radiation at Hanford 1944—1959. A. R. KEENE, General Electric Co., Hanford Laboratories, Richland, Washington.

Health physics at the Naval Research Laboratory. L. F. GARCIA, Naval Research Laboratory, Washington, D.C.

The significance of health physics evidence in the trial of a radiation personal injury action. HOWARD K. SHAPAR, Chief Counsel, Idaho Operations Office, U.S. Atomic Energy Commission, Idaho Falls, Idaho.

There has been considerable speculation concerning the evidentiary value of film badge and dosimeter readings and air-monitor charts in the trial of a personal injury action based on alleged radiation injury. The case of *Jackson E. McVey and H. E. Northway* vs. *Phillips Petroleum Company* in the United States District Court, Southern District of Texas, Houston Division, a case of first impression in the courts of this country, provides forceful confirmation of the necessity for the use of reliable dosimetry and related techniques and the maintenance of accurate health physics records and data. This case, popularly referred to as "the Kellogg case," arose out of an incident which allegedly occurred on 13 March, 1957, in the South Houston laboratory of the M. W. Kellogg Company. The plaintiffs maintained that they had incurred certain physical disabilities as a result of having been exposed to radioactive particles which had been released from irradiated pellets consisting of iridium-192 and aluminum. One of the major factual issues at trial was the extent of the radiation to which plaintiffs had or could have been exposed and the type and extent of physical injury that might reasonably be ascribed to such an exposure. There is a discussion of the prerequisites for introducing into evidence data obtained from film badges, self-reading dosimeters and air-monitor charts. Once the health physics data has been introduced into evidence, the interpretation of the data becomes the subject of expert testimony—that of the medical and health physics witnesses. The role of the health physics expert witness in drawing conclusions from the data is discussed. Unless the terminology of the trade can be meaningfully translated into clear and understandable equivalents, the value of the health physics data and records may be attenuated or lost.

The role of the health physicist on the environmental health team. R. O. WOLLAN, University of Minnesota, Minneapolis, Minn.

Radiological safety for underground nuclear explosions.* F. W. WILCOX, Radiological Safety Division, Health, Medicine and Safety Department, Reynolds Electrical and Engineering Co., Inc., Mercury, Nev.

The possibility of using nuclear detonation by industry in producing energy for various peaceful uses (power, excavation, heat, etc.) appears to be realistic. Consideration must be given to providing safeguards against radiation exposure or contamination from fission products and activated material.

Since the detonation of several nuclear devices at the Nevada Test Site, during operations Plumbbob and Hardtack-Phase II, mining and drilling operations have penetrated the resulting debris. Radiation levels as high as 150 mr/hr (γ) in the personnel work area and debris measuring greater than 25 rads/hr ($\beta + \gamma$) and 2×10^6 dis/min per 55 cm^2 (α) have been encountered as long as 7 months after the initial detonations.

Details of the procedures and experiences in providing the radiological and toxicological safety program are presented with exposure and contamination data.

Evaluation of the ventilation rate from the decay of filter paper air samples of thoron or radon daughters. K. S. SOMAYAJI, Atomic Energy Establishment, Bombay, India.

The inhalation hazard associated with the processing of thorium and uranium is mostly due to the daughter products of thoron and radon rather than to the parent gases themselves. With the provision of ventilation the equilibrium between the parent and the decay products is disturbed resulting in a marked reduction of the inhalation hazard. TSIVOGLOU *et al.* described a method of estimating the relative proportions of the various daughter products of radon in the atmosphere, based on an analysis of the shape of the α-decay curves of filter paper air samples.

This paper describes a simple method of obtaining the ventilation rate in a thoron field and the relative nonequilibrium among the decay products of thoron. It is indicated how the ratio of the α-activity after a delay of 150 min to that after a delay of 30 min can be used as a convenient parameter to do this.

Curves are given showing the variation of the potential α-energy values for thoron and radon daughters with the ventilation rate, which are useful when planning control by means of ventilation.

Distribution of specific radionuclides in the hydrosphere below KAPL. ALBERT G. FRIEND, ALBERT H. STORY, MICHAEL HOWELL, Radiological Health Research Activities, and Croswell Henderson, Water Supply and Pollution Control Research, Robert A. Taft Sanitary Engineering Center, Cincinnati, Ohio.

Waste materials containing nuclides have been and are being discharged from atomic energy installations and from facilities processing and using radioactive isotopes. With few exceptions, little information is available on the

*Work was performed under the auspices of the U.S. Atomic Energy Commission.

ultimate fate of these nuclides when released to a water environment. This study was initiated to gain a better insight into what happens to these nuclides when released to a stream.

γ-Spectra of selected samples of fish, filamentous algae, plankton, aquatic vascular plants, snails, bottom muds, water and KAPL effluent are shown at five stations on the Mohawk River in the vicinity of KAPL and two on the Hudson River. One station on the Mohawk River was located about 8 miles above the KAPL outfall, one in the outfall, and three below the outfall. One Hudson River station was located above and one below the point of entry of the Mohawk into the Hudson River.

Results show that the activity of the water, which can be attributed to individual isotopes, diminishes as the flow proceeds down-stream. Results of analyses show that, of the media tested in this environment, bottom muds are efficient at removal from the water phase of cesium–barium-137, zirconium–niobium-95, and strontium-90; filamentous algae are efficient for the removal of cesium–barium-137, zirconium–niobium-95, and cerium–praseodymium-141,144; fish are the most efficient for the removal of ruthenium–rhodium-106.

Artificial media, consisting of bagged tea, spinach and live algae, and ion-exchange columns, were also tested as possible monitoring devices. Results indicate that tea and algae are most promising, with tea showing less specificity of uptake than the algae.

Contributed Papers on Internal Radiation Hazards and Internal Dosimetry. Chairman: L. S. SKAGGS, Argonne Cancer Research Hospital, Chicago, Illinois.

The limitation of tracer doses of radioisotopes in man. EDWARD W. WEBSTER, ARTHUR K. SOLOMON and CHARLES V. ROBINSON, Harvard Medical School, Massachusetts General Hospital, Boston 14, Massachusetts.

Tracer doses of radioisotopes are administered to two groups of people—those who are sick and expect to derive some benefit from the radiation dose sustained in terms of diagnostic information; and those who are well but are acting as volunteers in research programs. The desirability and the philosophy of dose limitation to these different categories is discussed. Possible criteria for limiting the dose to volunteers are compared, particularly with dose limits already established for the general population and with those for individuals occupationally exposed. Comparisons are made between possible radiation hazards and other risks of living. The effect of such dose limitation on several important diagnostic isotope procedures is discussed. In summary, a policy which has gained acceptance in some medical schools is presented. This statement includes a table of critical organ dose for common diagnostic radioisotope procedures.

Studies on the behavior of inhaled fission products. ROBERT G. THOMAS and ROBERT H. WILSON, University of Rochester, Rochester 20, N.Y.

Rats and dogs were exposed to clouds of fission products released from the high-temperature meltdown of reactor fuel elements in the field. The primary goal was attainment of good lung deposition values which could be used in the initial step of the biological counterpart of reactor hazards evaluation. There was a total of seven releases with biological participation, three from aged and four from green fuel elements, and each under various conditions of lapse or inversion meteorological regime.

In the early releases, even with the animal sacrifice occurring shortly after exposure, the gastrointestinal (GI) tract values obtained were unrealistically high due primarily to licking. During the course of the Testing Program the GI tract-to-lung ratios in rats were reduced from >30 to <1 by altering such parameters as exposure cage construction, degree of consciousness of animals during exposure, methods of sacrifice, and time of entry into the network for pickup and sacrifice. In both rats and dogs the mean lower respiratory tract deposition of aged fission products was about 20 per cent and of green fission products was about 35 per cent of the estimated amount of material breathed. When the lung deposition values from all animals of one species are combined they exhibit a log-normal distribution pattern with a mean of 29 per cent for rats and 27 per cent for dogs.

Plutonium dioxide aerosol metabolism.* DONALD H. WILLARD, Hanford Laboratories, General Electric Co., Richland, Washington.

Adult beagle dogs were exposed to $Pu^{239}O_2$ aerosols of varying concentrations and particle sizes. From 50 to 70 per cent of the deposited dose was present in lungs immediately after exposure and the rest was divided between the nasal and oral passages, and the gastrointestinal tract.

Between 53 and 63 per cent of the total $Pu^{239}O_2$ deposited was present in the lung for 2 weeks after exposure. The insolubility of $Pu^{239}O_2$ in lungs will be demonstrated by the extremely small quantities which appeared in bone and other tissues. Early translocation of plutonium from lung to tracheobronchial lymph nodes was demonstrated. During the 2-week period an average of 51 per cent of the total $Pu^{239}O_2$ deposited was excreted in feces and only 0.36 per cent in urine. There was no obvious effect of particle size and magnitude of dose on distribution, translocation, and excretion of plutonium.

An acute inhalation dose of 50–150 μc $Pu^{239}O_2$ caused death in fourteen dogs within a 50–75-day post-exposure period. Beginning about 1 month after exposure there was an increased respiratory rate from about twenty per minute to about 180 respirations per minute prior to death. Lymphocyte values in the two groups of dogs receiving the highest doses of plutonium were significantly lower than control values for 16 weeks following exposure. Lymphocyte levels in the highest group decreased to about 20 per cent of control values.

An evaluation of the hazards of internally deposited zinc-65.* ROY C. THOMPSON and JOHN E. BALLOU, Hanford Laboratories, General Electric Co., Richland, Washington.

Data are presented on the accumulation of Zn^{65} in

* Work performed under Contract No. AT(45–1)–1350 between the Atomic Energy Commission and the General Electric Company.

various organs and tissues of the rat during over a year of continuous feeding of the radioisotope. Highest Zn^{65} concentrations were noted in hair, bone and prostate, the three tissues which did not attain steady-state levels of Zn^{65} during the feeding period. These results are compared with previously reported data on the retention of a single dose of Zn^{65} in animals, with data on the total-body retention of Zn^{65} in man, and with data on the ingestion and distribution of natural zinc in man. While differences are apparent between the acute and chronic exposure data, and between the small animal and human data, the general similarities are such that, taken together, they may be considered to quite closely define the parameters necessary for the evaluation of permissible body burdens and maximum permissible concentrations.

Estimation of dose from a single intake of radioactive material. W. S. SNYDER, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

There is a close relation between MPC values determined on the basis of chronic exposure and the total dose delivered by a single intake of a radionuclide. If the MPC, or equivalently the maximum permissible daily intake MPI, is determined so that the critical organ receives R rem/day after exposure at the MPC level for T years, then a single intake of the MPI will give a total dose of R rem throughout the following T years. This relation does not depend on the particular form of the retention law, i.e. it is valid whether the retention in the body is described by a power law or by any other form of retention law. The actual distribution of dose within the period of T years will depend on the radioactive half-life and the retention law of the particular radionuclide, and this will affect the estimate of the hazard to be attributed to a single intake. The ICRP in its Munich Statement [Report on Decisions at the 1959 Meeting of the International Commission on Radiological Protection, ICRP, *Health Phys.* **2,** 317 (1960)] has given recommendations concerning single intakes of radioactive material which are similar in spirit to the recommendation concerning emergency exposure to external irradiation. It is shown that the ICRP recommendations on single intake can be interpreted in terms of total dose by the use of the above-mentioned relation of MPC values and total dose. The biological data now available do not permit a definitive estimate of the relative hazard of a short-term vs. a long-term dose. Some of the relevant data are discussed.

Suggested MPC values for radioiodine isotopes in foods. J. H. HORTON, Savannah River Plant, E. I. Du Pont de Nemours and Company, Aiken, South Carolina.

The MPC values for I^{131} (chronic exposure) and I^{131}, I^{132}, I^{133} and I^{135} (acute exposure) were calculated for eight food groups. The dietary, anatomical and physiological characteristics of twenty categories of people were considered so that the calculated values are applicable to important segments of the American population.

Evaluation of errors affecting enriched uranium exposure estimates. P. R. AMMANN and L. E. BURKHART, Y-12 Plant,* Oak Ridge, Tennessee.

A general method for estimating internal exposure to enriched uranium by means of urinalysis techniques is described by equation (1):

$$D = KVN \qquad (1)$$

where D = estimated internal exposure (mrem/day, or dis/min per day);
K = a constant comprised of factors describing the assumed path of uranium through the body;
N = net α-activity concentration in urine sample as determined by urinalysis;
V = void rate (either measured or assumed).

An investigation of the effect of errors and biases contained in the void rate, V, and the urine activity, N, upon the calculated exposure, D, was conducted. The following were included:

(1) An experimental study was carried out in which three independent AEC bio-analysis laboratories, employing different urinalysis methods, analyzed a series of common urine samples. The range of enriched uranium activity in the samples varied between zero and three times the maximum permissible concentration. A statistical analysis of the data revealed:

(a) that sources of bias (inaccuracy) were present in all three methods;
(b) that the major sources of errors in analytical results were incurred not in the statistics of radiation counting, as usually assumed, but in variability of equipment and individual analytical techniques.

(2) A study of the relationship of void rate and urine activity concentration to calculated exposure was made using existing exposure histories of nearly 800 employees covering a period of $2\frac{1}{2}$ years. The data (approximately 60,000 points) were summarized utilizing a digital computer (an IBM 704).

By means of a cathode ray tube–camera ancillary to the computer, dose, void rate, and urine activity were plotted vs. time for the individual employees, thereby producing visual summaries of entire exposure histories. In addition, histograms of these same variables were graphically represented, and parameters of special interest, such as the average exposure for the $2\frac{1}{2}$-year span, were calculated.

A statistical correlation of calculated exposure with void rate and urine activity was made utilizing the above described data. It was shown that both high void rates and high urine activities bias the calculated exposure such that a high dose is yielded. Conversely, low values result in a correspondingly biased exposure. Empirical relationships were computed which allow these effects to be compensated.

(3) Studies of the electrodeposition of uranium from untreated urine showed that:

(a) Electrodeposition efficiency varied with urine acidity and with uranium concentration in the range of 0.05 to 0.5 μg/l. (the maximum permissible concentration for enriched uranium is approximately 0.25 μg/l.).

* Operated by Union Carbide Nuclear Company, Division of Union Carbide Corporation, for the U.S. Atomic Energy Commission.

(b) A significant fraction of uranium in solution (concentration range of 0.2 to 5.0 μg/l.) adsorbed onto surfaces of glass containers within several hours at a pH of 6, but no adsorption was detected at a pH of less than 4 over a period of 5 days.

New biological exposure constants. J. C. COUCHMAN, Convair, Fort Worth, Texas.

The fission products nuclear data given in the literature have been compiled on IBM cards for the IBM-704. They have been operated on using biological exposure equations to obtain $\Sigma E_{eff}(\text{RBE})N$ for 168 fission products and to compute their rem per curie (RC's) in each of twelve body organs. Practical examples of the uses of the tables are presented.

While preparing the tables, it became evident that there are apparent voids in some of the basic biological parameters such as the T_{biol}'s the f_a's and the f_w's. To supply what is needed will require thorough screening of biological experiment reports and new experimentation. Pertinent recommendations are made.

All fission products with half-lives greater than 1 min that are the first in an isobaric chain have been considered along with all their daughter products. $\Sigma E_{eff}(\text{RBE})N$ was computed for each isotope for the total body, bone, kidney, muscle, thyroid, lung (soluble), lung (insoluble), GI (insoluble), spleen, liver, ovary and testes using the following equation:

$$\Sigma E_{eff}(\text{RBE})N = N\sum_{\beta} 0.33 f_{\beta_i} E_{\beta_i}\left(1 - \frac{\sqrt{(Z_i)}}{43}\right) \times \left(1 + \frac{\sqrt{E_{\beta_i}}}{4}\right) + \sum_{\gamma} E_{\gamma_i} f_{\gamma_i}\{1 - \exp[-(\mu - \sigma_s)r]\} \times \text{MeV/dis}$$

Values of f_{β_i}, E_{β_i}, Z_i, E_{γ_i} and f_{γ_i} were taken from the above references. The effective radii were the values recommended for standard man, and the values of $(\mu - \sigma_s)$ were taken from the energy absorption cross section curve for tissue. Two computations of $\Sigma E_{eff}(\text{RBE})N$ were made for bone (one with $N = 1$ and one with $N = 5$ for non-uniform β-distribution).

The value of RC (rem per curie in body organ) was computed for each isotope in each body organ from

$$\text{RC} = \frac{7.38 \times 10^7 T_{eff}}{M} \Sigma E_{eff}(\text{RBE})N \times \left\{1 - \exp\left[\frac{-0.693(70y)}{T_{eff}}\right]\right\}$$

where T_{eff} was computed by: $T_{eff} = \dfrac{T_r T_b}{T_r + T_b}$ using the biological half-lives suggested by Dr. K. Z. MORGAN.

Uranium oxide smoke from exploding wires. FRANK G. KARIORIS, Physics Department, Marquette University, Milwaukee, Wisconsin, and BIRNEY R. FISH, Health Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

This is a preliminary report describing an exploding wire apparatus suitable for the production of metallic oxide smokes for use in animal inhalation or other studies. Uranium oxide particulates with 0.034 μ average diameter have been produced with little evidence of chaining or agglomeration. Approximately 90 per cent of the metallic sample can be converted to the oxide smoke with good reproducibility of yield.

The apparatus consists of a 20 μF capacitor bank which may be charged to 20 kV and discharged through a measured mass of foil or wire clamped between the electrodes of an explosion chamber. The high-energy surge explodes more than 20 mg of uranium or other metals producing a metallic oxide particulate aerosol when the explosion chamber is filled with air. Provisions are included to sample, collect and dilute the smoke, and to decontaminate and control the atmosphere of the explosion chamber.

Contributed Papers on Environmental Hazards of Radiation and Reactor Accidents. Chairman: J. F. SOMMERS, Phillips Petroleum Co., Idaho Falls, Idaho.

Problems associated with radiation incidents. DEAN D. MEYER, Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

Incidents involving radioactive materials vary widely from minor ones, such as the use of a contaminated tool in an office area, to such very extensive and serious ones as would result from a fire or reactor incident releasing a potentially contaminated cloud of airborne material. This discussion will cover plans for handling incidents in the minor and moderately extensive range, and will include the problem of permissible levels to be used in decontamination of uncontrolled areas.

Health physics considerations in the ICPP critical incident of 16 October, 1959.* J. W. MCCASLIN, Phillips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho.

A critical incident occurred in the ICPP on 16 October, 1959. The incident resulted from the unexpected collection in a waste vessel of a concentrated solution of about 34 kg of U^{235}. It is estimated that 4×10^{19} fissions took place in the vessel. The reaction was shielded by about 4 ft of concrete from any working area and by considerable additional concrete and distance from any area occupied at the time.

Nineteen men were in the plant area, twelve were in the immediate building. Despite the high level of reaction and, consequently, the high resulting radioactivity, several factors, in the author's opinion, aided in holding personnel doses of radiation to relatively low figures. These factors were: (1) the remote location of the vessel; (2) the heavy shielding; (3) good radiation monitoring; (4) prompt response by personnel to the resulting alarms; (5) the small number of men in the plant. Severe meteorological adversity added materially to the dosages.

* Work performed under USAEC auspices.

Document IDO-10035 is given as a reference in regard to the basic accident. Several minor points in improved alarms, additional training, and detailed procedures are discussed.

Airborne argon-41 at NRL reactor. MICHAEL P. DOYLE, Naval Research Laboratory, Washington, D.C.

The swimming-pool-type reactor at the U.S. Naval Research Laboratory is operated at a power of 100 kW. At a given ventilation rate, the concentration of airborne A^{41} inside the reactor building was found to reach a value of 80 per cent MPC. The A^{41} measurements involved collecting an air sample in two 100 cm^3 chambers having thin end windows and counting the sample with an end window ("pancake") G.–M. tube. This system was calibrated by using known concentration of A^{41}.

The potential sources of the airborne A^{41} are the beam ports, thermal column, pneumatic rabbit system, and pool water. Radiological surveys were carried out to determine the principal source(s). The A^{41} concentration inside the building was measured while all gases from one of the potential sources were prevented from contaminating the air. A comparison of the results with those obtained under normal operating conditions would then indicate the significance of the contribution from that source. These surveys indicated that air from the voids of the shield (thermal column and beam ports) accounted for 75 per cent of the airborne A^{41} in the building. The remainder was attributed to the air dissolved in the pool water. There was no measurable contribution from the pneumatic rabbit system.

Elimination of the contribution from the principal sources (thermal column and beam ports) reduced the A^{41} concentration to less than 15 per cent MPC.

The environs monitoring program for the Dresden Nuclear Power Station. JOHN H. HUGHES, Dresden Nuclear Power Station, Joliet, Illinois.

The environs of the Dresden Nuclear Power Station are monitored to establish the preoperational and operational radiation levels in the vicinity of the plant. This program was initiated a year prior to the start-up of the power station to comply with public health and safety regulations, and in the general interest of public relations. It was formulated to give the necessary information to evaluate the overall effectiveness of the plant's design and method of operation to minimize and control the release of radioactive materials during operation with the utilization of the minimum amount of manpower and capital investment.

Dresden Nuclear Power Station, owned by Commonwealth Edison Company, is located approximately 50 miles southwest of Chicago, Illinois, at the confluence of the Des Plaines, Kankakee, and Illinois Rivers. The facilities are constructed on a 950-acre tract which affords a minimum of $\frac{1}{2}$ mile to the nearest unrestricted area. The reactor, a dual-cycle boiling-water type fueled with 65 tons of enriched uranium dioxide, has a thermal rating of 626 MW. The gross electrical capacity of the turbine-generator is 192 MW.

The radioactive gas and liquid waste processing systems are designed to permit continuous full power operation of the plant. Self-imposed design and operating criteria established for Dresden limit the normal release of radioactive gases and the resulting exposure to the inhabitants to less than one-tenth (actually 2–4 per cent) of the annual non-occupational permissible limit as defined in AEC Title 10, CFR, Part 20. Similarily, release of radioactive liquid wastes to the Illinois River are limited to one-tenth of the drinking water limit. In addition, the State of Illinois has defined certain limits.

The environs monitoring program consists of the collection of data and samples, analysis of the latter, and the evaluation and correlation of the results. Air, water, vegetation and animal samples are gathered on varying frequencies for the determination of the gross α, β–γ, β-decay, and certain specific radioisotopic content. Airborne particulate radioactivity is collected by filter paper method and analyzed weekly. Vegetation typical of the area are ashed and analyzed for β-activity and strontium-90. Well and surface waters are analyzed for α- and β-activity and uranium, potassium-40, strontium-90, and fission products. Silt, plankton and slime supplement surface water samples. Milk and fish are obtained quarterly for α- and β-activity, strontium-90, and fission product analyses. γ-Emitting radiogases are determined by means of paired ionization chambers (0–10 mr). The annual program includes 1827 analyses for total activity; 213 for specific isotopes, and a total of 1623 samples.

The program is conducted by an outside contractor, who is responsible for collection, shipment, analysis, calculation and evaluation of all phases. The contractor was selected by competitive bidding from a group of qualified firms. Results of the first year's operation of the program are presently being compiled and evaluated prior to submission of the annual report. (These results will be presented in detail at the June meeting.)

The radiation detection and monitoring system for the Piqua Organic Moderator Reactor. A. R. PICCOT, Atomics International, Canoga Park, California.

A description is given of the radiation detection and Monitoring system for the 45.5 MW(th) Organic Moderated Reactor presently under construction in the City of Piqua, Ohio. The system provides all the necessary radiation detection instrumentation and annunciations or protective action to insure that operating personnel and the general public are protected against radiological hazards arising from operation of the plant. The systems were designed to insure compliance with the regulations specified in 10 CFR 20. Included in the system are continuous gaseous and liquid effluent monitors, continuous area radiation and airborne activity monitors and a normal complement of health physics survey and counting room instruments. The need for the individual instruments, their required sensitivity and application are briefly described. The entire radiation monitoring system utilizes only standard commercially available instruments.

Radiation effects of a 10-MW reactor on environs of AFP 67. J. M. SELBY, C. A. WILLIS, B. M. BOWEN and J. H. EDGERTON, Lockheed Nuclear Products, Lockheed Aircraft Corporation, Georgia Division, Marietta, Georgia.

The Radiation Effects Reactor has been operated for approximately 600 MW hr. During that time, studies have been made to determine the radiation and activation effects on the environs of AFP 67. Soil, air and water samples have been collected to determine the amount of activation and the resulting buildup of radioactive materials. Instrument surveys and foil measurements have been made to determine radiation levels. The effects of high γ and neutron exposures on plant life have been observed as part of an ecology program performed in conjunction with Emory University. This paper is a preliminary report of these studies.

Airborne fission products at the CP-5 reactor. ROBERT V. WHEELER and JOHN J. HARTIG, Argonne National Laboratory, Lemont, Illinois.

Air samples taken in the reactor room of the Argonne CP-5 installation during the fall of 1959 indicated abnormal concentrations of radioactivity. A study to determine the type, amount and source of this activity was undertaken. The results of the study indicated that the activity was fission products with cesium-138 being prominent. The release of the activity was traced to the reactor helium system which was contaminated. The contamination may have resulted from several sources. Investigation indicated that the most probable source was pinhole defects in the fuel cladding.

Radiation safety aspects of the EBWR vessel modifications. JOHN J. HARTIG and HENRY A. TILL, Argonne National Laboratory, Lemont, Illinois.

Operation of the Experimental Boiling Water Reactor demonstrated that the reactor is capable of powers above the design level of 20 MW (thermal), the limitations being the heat-removal facilities. The plant was shut down July 1959 and work to increase these facilities begun. Much of the work was performed in areas of high background and some within the reactor vessel itself.

Fuel handling during the core unloading was routine. The loading of fuel into the storage pool was monitored for neutron multiplication. None was noted. Disassembly of the control rod drives was complicated by an active sludge (Co^{58} and Co^{60}) on the reactor bottom, producing backgrounds as high as 10 r/hr. The control rods themselves were cut into three sections by a pneumatic saw under 10 ft of water and then removed utilizing the spent fuel coffin. Saw chips, collected in a basin, were active to 100 r/hr. Modifications within the vessel were performed in backgrounds reduced to 50 mr/hr by a lead shield positioned above the shroud. Vessel decontamination was affected initially by hand washings and later with a high-pressure hose.

Airborne activity sampling indicated that at no time were personnel exposed to MPC or above concentrations. Urine samples all had negative results. The maximum external exposure received was 3.3 r over a 6 month period.

It has been shown that revisions on a power reactor vessel may be conducted with acceptable personnel exposures.

Health physics for the *N.S. Savannah*. ERNEST P. RESNER, States Marine Lines, c/o New York Shipbuilding Corp., Camden, New Jersey.

This paper presents the role of health physics and environmental considerations in the operation of the *N.S. Savannah*. A brief description of the vessel includes a review of the inherent safety features which were designed to make this ship acceptable in domestic and foreign ports. Maximum safety for the ship, its passengers, complement and environment implies that a maximum effort in health physics be expended. The scope of the shipboard program involves routine and special surveys, air analysis, radiochemistry, personnel monitoring and waste management.

A pre-operational site survey has been established in conjunction with the shipyard, the States of Pennsylvania, New Jersey and Delaware, and with the U.S. Public Health Service. The off-site environmental analysis has included the consequences of the release of radioactivity to the atmosphere and the Delaware River.

Radiological health control is further concerned in support facilities designed to service the reactor system. For the servicing of the *Savannah*, a barge has been designed to facilitate major reactor system maintenance, refueling operations and management of wastes.

Dispersion of radiocontaminants in an estuary. F. L. PARKER, G. D. SCHMIDT,* W. B. COTTRELL and L. A. MANN, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

The dispersion of radioactive contaminants in an estuary is more complex than the dispersion in a flowing fresh-water stream due to tidal cycling in the estuary. Many approximate analytical solutions are available for the hydraulic dispersion but a model study is more accurate. The model study, however, ignores radioactive decay, and the absorption and adsorption of the nuclides on the sediments and the biota. An evaluation of the maximum credible accident to the *N.S. Savannah* during startup in the Delaware River shows that hazardous conditions will not exist. The evaluation was based on model studies by the U.S. Army Engineer Waterways Experiment Station and the nuclear decay and sorptions of the nuclides. The maximum credible accident assumes 600 days operation at 69 MW and that 1003 of the fuel rods fail. It also assumes that only a percentage of the isotopes will escape from the containment vessel and only at a leakage rate of 0.2 per cent of the free volume of the containment vessel per day. Exposures from swimming and boating in the water or drinking the water, have been evaluated and show that the maximum hazard is that the maximum concentration in the river will be about twice

* Public Health Service.

the maximum permissible concentration for continuous occupational exposure for a few days.

Health physics aspects of the Westinghouse testing reactor fuel element melt-down. ROBERT J. CATLIN and D. C. COLLINS, Health Physics Section, Westinghouse Testing Reactor, Pittsburgh, Pennsylvania.

On 3 April, 1960, the Westinghouse Testing Reactor experienced a fuel element melt-down resulting in the release of large amounts of fission products to the primary coolant loop of the reactor. The high radiation levels which resulted caused initial plant evacuation and hampered subsequent decontamination and system clean-up. This paper reviews the radiation hazards involved and the techniques and equipment employed for return of the reactor to operating condition. Review is also made of the several radioactive booby traps encountered and of techniques planned to eliminate these in future operations.

The Lockport incident—accidental radiation exposure from radar equipment. HERBERT MERMAGEN, CARL L. HANSEN, JR. and JOE W. HOWLAND, University of Rochester, Rochester, New York.

Contributed Papers on Fallout and General Topics in Health Physics. Chairman: A. L. CARSTEN, Brookhaven National Laboratory.

Current status of fallout strontium-90. ARTHUR R. SCHULERT, Lamont Geological Observatory (Columbia University), Palisades, New York.

Strontium-90 is generally regarded to be the greatest long-term somatic hazard among the radioisotopes of nuclear fission. As a result of the 1½ year moratorium of atomic bomb testing by the three principal nuclear powers, the concentration of strontium-90 in foodstuffs is decreasing and consequently the rate of incorporation into growing bones is now less than at the peak levels of the summer of 1959.

Radiochemical analyses of the β-emitting strontium-90 are a long and rather expensive procedure. Thus both time and money limit the quantity of analyses; furthermore, results on a given set of samples are not quickly available. Fortunately the problem never became critical during atomic testing since only in rare sporadic instances did a single food from a local area exceed assigned MPC values, and no human bones, among about 8000 samples from around the world, have contained more than a few per cent of the strontium-90 MPC. Thus the precise radiochemical procedures on a limited number of selected samples have been adequate for determining the environmental contamination, its variation with geography and diet, and its change in time. It is clear, however, that these conventional radiochemical procedures would be inadequate for the extensive monitoring required should widespread contamination at and above MPC levels occur.

Three useful approaches to quicker approximate strontium-90 methods have been considered: (a) β-counting of crude, partially-purified samples; (b) assay of the bremsstrahlung produced by strontium-90; and (c) estimations of the strontium-90 from the γ-emitters from fission which can be quickly counted. Procedure (a) may still not be sufficiently rapid for quick large-scale monitoring. Other activity may mask the strontium-90 bremsstrahlung limiting the value of procedure (b). Procedure (c) would be complex, but if correlative data were in hand relating the activities as a function of the various parameters, this method may prove the best for quick strontium-90 estimation in foods.

A method of evaluating fallout radiation protection. MARLOW STANGLER, Radiological Physicist, Office of Civil and Defense Mobilization, Battle Creek, Michigan.

In the event of nuclear attack, shelter from fallout offers the best single nonmilitary defense measure for saving the greatest number of people. Existing large buildings afford varying degrees of fallout protection. An analytical method was needed to adequately assess the potential existing fallout shelter.

To provide basic data on penetration of nuclear radiation for practical application to evaluate the fallout protection, a combined theoretical and experimental approach was taken. The theoretical work has provided a method for calculation of the penetration but is complicated for extensive use involving complex structural geometries. Simplifying assumptions were made to develop a practical method but the degree of accuracy resulting from the assumptions needed experimental verification.

An experimental means for simulating the fallout radiation field was developed, consisting of a 200 c Co^{60} source circulating through prelayed tubing. Radiation reaching points of interest within an actual structure was integrated by pocket dosimeters, and normalized dose rates for a specific fallout density were determined.

This research program resulted in development of an analytical method that considers the interrelationship of barrier and geometry shielding for evaluating the radiation protection in structures. Barrier shielding is calculated using the most appropriate of three cases considering the relative positions of the barrier, the contamination and the detector, and the appropriate mass thickness. Geometry shielding calculations consider the detector position, the area, shape and height of the building. Corrections are made for factors such as apertures, skyshine and adjacent buildings. This method has been used in several fallout shelter surveys conducted by OCDM.

The prediction of residual γ-radiation levels from fallout after a multi-weapon nuclear attack. WAYNE M. LOWDER, LEONARD R. SOLON, and ALFRED J. BRESLIN, USAEC, New York Operations Office, New York.

The problem of predicting γ-radiation dose rates as a function of time and total doses to exposed individuals in the superimposed fallout patterns from a number of nuclear explosions is a complex one. Two approaches are described which allow a reasonable prediction to be made

in a practical situation, within the limitations of our present knowledge of the nature, deposition and decay of the radioactive products of such explosions. The mathematics involved are relatively simple, and calculations can be performed in the field (i.e. a fallout shelter) with the aid of a few tables or graphs.

The approximations involved in the methods considered as well as deficiencies in our knowledge of the relevant properties of nuclear weapons detonations are discussed, and it is concluded that the necessities of field prediction in an actual emergency will render such an approach of considerable utility.

Fallout transport as deduced from measurement of γ-emitting radioactivity in air. PHILIP F. GUSTAFSON, Argonne National Laboratory, Lemont, Illinois.

Scintillation spectrometry was used to determine the concentration of fission and activation products in air filters operated at ground level. The activity ratios of specific radionuclides (Ru^{106}, Ce^{144}, Cs^{137} and Zr^{95}) as well as W^{185} and Rh^{102} tracers have been utilized to assign nuclear debris to particular test series during the past several years. The short stratospheric residence time of much of the test debris will be discussed in some detail. (This work was performed under the auspices of the U.S. Atomic Energy Commission.)

Deposition velocities of airborne fission product nuclides. Dr. R. E. FIELDS, Convair, Fort Worth, Texas.

Reactor hazards analysis usually includes the requirements for estimating the geographical extent and radioactive intensity of ground deposition of known air concentrations. One of the important parameters involved is the so-called "deposition velocity, V_g" or ratio of concentration on the collecting surface to the time integral of concentration in the air. Values reported in the literature for various aerosols (including SO_2, smoke, fluorescent pigments, lycopodium spores, I^{131}, and nuclear bomb debris) range from 10^{-2} to 10^{-3} m/sec.

This paper reports the results of a large number of field measurements made with fission products released by melting irradiated reactor fuel elements. V_g's for several of the biologically most hazardous nuclides have been determined. For instance, a V_g for cesium-137 on to gummed paper of 0.010 ± 0.006, and on to soil of 0.004 ± 0.002 m/sec was obtained in a series of field tests conducted in 1958, over a sage-brush-covered, dry desert. In a second series of field tests conducted in 1959 over a barren, wet desert, a value (preliminary) of 0.03 m/sec on to gummed paper was obtained. Values of 0.11 and 0.03 m/sec were obtained for iodine-131 on gummed paper in the first and second series, respectively.

Differences in values obtained under various conditions of terrain and meteorology will be discussed, e.g. differences which may be attributable to variations of turbulence and dustiness of the atmosphere, distance of the collector from the source, etc.

The measurement of air flow through high efficiency filters C. L. LINDEKEN, D. N. MONTAN and E. L BEARD, Lawrence Radiation Laboratory, Livermore, California.

A need existed at the Lawrence Radiation Laboratory, Livermore for a continuous measurement of air flow through high-efficiency filters as a criterion for filter changing. These filters are changed when the dust load prevents the flow of sufficient air to provide adequate ventilation. Heretofore, however, the estimation of flow has been rather qualitative. The results may be a compromise of safety if the filter change is delayed, or an economic loss if the change is made while the filter still has useful service. Since a large number of filters were involved, consideration of cost as well as performance was of importance in selecting a flow measuring system.

Choice of flow measuring system

The pitot tube was chosen for use on the basis of cost ease of installation, and minimum interference with air flow. The preferred location of the flow sensing unit is on the clean side of the filter. However, due to space limitations or other circumstances, installation must often be made before the filter in areas where the duct is contaminated. Under these circumstances, installation of orifice plates or venturi meters represent a hazard in itself since the duct work must be separated. In contrast, pitot tubes can be inserted through a single $\frac{5}{16}$ in. drilled opening.

Determination of velocity pressure

A commercially available plastic rotameter was used to measure velocity pressure. The flow, resulting from impact plus static pressure, enters at the bottom of the variable area tube. As this pressure always exceeds the opposing static pressure when air is flowing normally in the duct, the float will always read positive. Although not intended for use in the manner described, this procedure serves to subtract the static pressure and indicate velocity pressure only. Velocity pressures as low as 0.01 in. of water that can be easily detected.

Calibration of pitot tubes and field use accuracy

For 2 in. duct systems, pitot readings were compared with a reference venturi meter. In larger ducts comparison was made with a velometer and a reference pitot tube. Although the accuracy obtained with the system as routinely installed ($\pm$10–15 per cent) is less than desired, this accuracy is sufficient for the purpose intended. Once a clean filter is installed, we are interested in noting decreases in flow caused by dust buildup. Thus, relative rather than absolute measurements are required. Conditions governing the flow pattern within the duct which account for measurement errors normally remain constant during the life of the filter.

Recommended use of the system

(1) With a new filter installed, the manifold damper leading to the filter is adjusted to give the desired flow. Rotameter reading is noted.

(2) As dust accumulates on the filter, the damper is opened to increase the flow to the original rate.

(3) When it is impossible to achieve proper flow with the damper wide open, the filter is replaced.

Fission product release from overheated uranium—A laboratory study. R. K. HILLIARD, C. E. LINDEROTH and A. J. SCOTT, General Electric Company, Hanford Product Operation, Richland, Washington.

The nature and amount of radioactive material released during a reactor accident are of importance to reactor hazards evaluations, design of containment facilities and establishment of operating criteria. A laboratory study is currently underway at Hanford to provide experimental data which will be pertinent to analyses of accidental release under a wide range of reactor conditions.

The investigation is designed to determine the effects of temperature, time of heating, atmosphere in which heated, irradiation level and specimen size on release of eight key fission-product elements from irradiated normal uranium.

Preliminary tests investigated the oxidation and ignition behavior of unirradiated uranium in air and steam. Small uranium cylinders were then irradiated to 2×10^{14} nvt and heated out-of-reactor under carefully controlled conditions and the fractions of the fission product isotopes leaving the specimen were determined. Temperatures up to 1440°C for as long as 2 hr in atmospheres of air, steam and helium were measured; release in air was two to ten times greater than in steam, while in helium the release was very low. Iodine, tellurium and xenon were released in proportion to the amount of uranium oxidized. Cesium and ruthenium were found to be semi-volatile. The release of strontium, barium and zirconium was found to be less than 0.2 per cent under all conditions tested.

Efficiencies of several filter media in retaining volatilized fission products were measured. Release during unsteady temperature conditions were compared with release during isothermal tests. Diffusion coefficients were measured for several isotopes diffusing in metallic uranium at 1000, 1100 and 1200°C.

Tests are currently being performed to determine the influence of high irradiation level and specimen size on the fraction of fission products released from heated uranium.

Leaks in standard sealed sources. R. J. CLOUTIER and MARSHALL BRUCER, Oak Ridge Institute of Nuclear Studies, Oak Ridge, Tennessee.

Teletherapy sources encapsulated in the United States and Canada from 1954 to 1959 were generally of the standard, single capsule, sealed type. One feature of the standard capsule was believed to be its tight seal and its ability to maintain this integrity. Late in 1958, an Oak Ridge Institute of Nuclear Studies standard capsule source, encapsulated 1 year earlier with Co^{60} metal pellets, developed a leak. This leak resulted in moderate Co^{60} contamination before being detected. Subsequent investigation of other sealed sources in the United States revealed that some had developed leaks. A review of the possible causes of leakage, and of the standard capsule design, fabrication, sealing, and initial and subsequent leak testing, has resulted in recommended changes to reduce the possibility of future capsule failures. The incidence of failures already encountered, however, has made apparent the importance of routine leak testing of encapsulated sources.

A Monte Carlo code for calculating γ-ray attenuation. H. B. ELDRIDGE and R. H. RITCHIE, Oak Ridge National Laboratory,* Oak Ridge, Tennessee.

A program developed for the IBM-704 computer calculates by stochastic methods the distribution in energy, angle and space of γ-ray flux, and the distribution in angle and space of the physical dose from a source distribution that may be extended in both energy and space. The method consists of following the histories of individual photons. When a photon crosses a predetermined boundary one modifies the flux and dose at that point in phase space so as to include the contribution of that photon. Instead of allowing absorption to take place when collisions are experienced, a statistical weight is assigned to the photon emerging from a collision. An optional biasing scheme is available whereby one may focus attention on deep penetration. The program has been arranged so as to be adaptable to various complex recording geometries in heterogenous media; however, present investigations have been restricted to a recording geometry of concentric spherical surfaces about the origin in an infinite homogenous medium. Some computations using this program will be compared with the results of the moments method.

Calculations of weapons radiation penetration in air. R. L. FRENCH and M. B. WELLS, Convair, Fort Worth, Texas.

The spatial, angular and energy distributions of the initial radiation from a nuclear weapon were calculated on a theoretical basis. The radiation components considered were fission neutrons and prompt fission, fission product and air-capture γ-radiations. The principal simplifying assumptions employed in the theoretical model were neglect of: (1) self-attenuation in the nuclear weapon; (2) blast effect on air density; and (3) ground effects. Comparisons were made with measured data which clearly demonstrate that the spatial, angular and energy distributions of the initial radiations from a nuclear weapon are calculable.

Determination of radium-226 and thorium-230 in mill effluents. EARL R. EBERSOLE, ALAN HARBERTSON, J. KENNETH FLYGARE, JR. and CLAUDE W. SILL, USAEC, Idaho Operations Office, Idaho Falls, Idaho.

Title 10, Part 20 of the Code of Federal Regulations requires effluents from mills processing uranium ores to contain less than 4×10^{-9} μc/ml of radium-226 before they can be released to uncontrolled environments. To ensure compliance with these regulations, analytical procedures are required that are capable of detecting one-tenth of this quantity in the types of samples to be encountered. Procedures involving counting of barium

* Operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

sulfate carrier for radium-226 presently in the literature are inadequate when applied to mill effluent samples. Non-equilibrium ratios of thorium-230 and radium-223 isotopes present in mill effluents require decontamination factors of 10^4 for thorium and a correction method for radium-223 in order to determine radium-226 in the sample. The nature of the non-equilibrium problem is discussed and alpha energy spectra presented to substantiate the problems encountered with mill effluents.

A method of analysis using barium sulfate carrier precipitated from an alkaline solution of the sodium salt of ethylenediaminetetracetic acid (EDTA) which gives sufficient decontamination from thorium isotopes, and a method of correcting for radium-223 activity in the presence of radium-226 is given in the paper. The method described uses only conventional laboratory equipment and an α-counting instrument and is suitable for large numbers of routine samples.

α-Air monitoring with α/β ratio compensation for natural interference. C. L. LINDEKEN and D. N. MONTAN, Lawrence Radiation Laboratory, Livermore, California.

The α/β ratio α-air monitor is based upon the relative constancy between the α- and β-activities of filter-collected natural airborne activity and the increase in this ratio produced by the accidental release of long-lived α-contamination.

While the α/β ratio is independent of the concentration of natural activity, the amount of change in the ratio produced by a given quantity of α-contamination is inversely related to the concentration of natural activity. For example, 100 dis/min of plutonium produces a 100 per cent change in the ratio if the natural α-activity present is 100 dis/min. However, if the natural α-activity present is 1000 dis/min, the change in the ratio is only 10 per cent. Thus, the sensitivity of any method based on direct observation of the α/β ratio is variable depending on the concentration of natural background.

This paper describes an approach using the α/β method designed to improve the response to α-contamination.

The α/β ratio monitor assembled for evaluation (Nuclear Measurement Corp. AM3A moving tape assembly and PC21 gas flow proportional chamber) permits detection of activity concurrent with collection. Either moving or fixed filter collection can be provided as desired. Both α-and β-activities are detected simultaneously with the same detector and are fed separately to a dual channel linear count-rate meter equipped with a difference amplifier.

Assuming that β-airborne contamination does not occur simultaneously with an accidental release of α-activity, the total α-activity observed is the sum of the long-lived α-activity plus the natural α-activity while the β-activity is attributed to natural activity only.

Therefore, since $\Sigma\alpha = \alpha_{LL} + \alpha_{natural}$

and letting $\frac{\alpha}{\beta} = \rho$

where ρ = observed ratio taken in an area free of long lived contamination

then $\alpha_{LL} = \Sigma\alpha - \rho\beta$

A variable attenuator is installed between the output of the beta channel and the input of the difference amplifier. Then by attenuating the beta signal by the value ρ, the difference amplifier indicates long-lived activity directly.

Our experience with the α/β ratio monitor using this modification is discussed. Data on instrument stability and sensitivity are presented.

Dating the irradiation and cooling times of unknown, mixed fission product activity. ALAN A. JARRETT, Atomics International, A Division of North American Aviation, Inc., Canoga Park, California.

An essential part of the safety program associated with the use of radioactive materials is the routine determination of contamination on surface and in air and water. These determinations are made by organizations utilizing or generating radioactivity as well as a variety of public agencies concerned with surveillance of the environment.

When unexpected activity is detected and identified, it may be associated with a specific operation. Considerable complication is introduced, however, when the activity is determined to be mixed fission products. This is particularly true in an area which may have more than one source of mixed fission product activity. The sources may include reactor fuel processing, hot cell operations, reactors or critical experiments. An additional complication is the possibility that the mixed fission product activity may be the result of nuclear weapons testing.

As an aid in determining the origin of mixed fission product activity, an experimental method has been developed which dates the irradiation and cooling time of an unknown mixture of mixed fission products. The method may also be applicable to the identification of a specific fuel element failure in a nuclear reactor if the loading and operation has resulted in a variety of irradiation times within the reactor.

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SYMPOSIUM ON THE EFFECT OF THE RECOMMENDATIONS OF THE NATIONAL COMMITTEE ON RADIATION PROTECTION (NCRP) ON NATIONAL LIFE*

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INTRODUCTION

THE National Committee on Radiation Protection was founded because, during the period of the 1920's it was recognized, as the result of misuse of radium and a number of serious injuries incurred in the use of X-rays, that acceptable standards for X-ray and internally deposited radioactive material must be established. This committee co-operated with the International Commission on Radiological Protection in establishing a rather detailed set of practices. These practices were based on the best evaluation of data available at the time. The acceptance of these practices was largely on a semi-voluntary basis. The problem of radiation protection had not yet become of sufficient magnitude to attract a great deal of attention from national governments.

Within the few years since the invention of the nuclear reactor, radiological protection has gone from a health problem affecting a few industrial workers or a small number of physicians, dentists, physicists, radiochemists, cyclotron operators, X-ray machine operators and other professional people, to a problem which is the concern of every inhabitant of the globe.

It is the purpose of this symposium to reflect the opinions of the four speakers who approach this problem from different standpoints. We are interested in the risks which can be expected to be taken and the benefits to be received from the use of nuclear energy. For example, nuclear technology employs many thousands of people. It may be expected that it will employ many more thousands in the future. It is obviously to the advantage of Labor and the employee that these jobs be provided. On the other hand, it is also greatly to their interest that a proper balance be struck between the hazards of this new kind of employment and the economic benefits therefrom.

Ionizing radiation has been used by the medical profession in therapy and diagnosis for many years. Since it falls within the purview of the medical profession to use radiation as well as drugs and other treatment in such a way as is currently being practised, the medical profession would use standards other than the occupational exposure regulations in the diagnosis and the treatment of patients. On the other hand, there remains an obligation to the patient and to the public to maximize the benefit to be received from the use of radiation and to minimize its deleterious effects.

The relationship between the number of incidents in which there is radiation injury or overexposure, the amount of plant investment and operating costs which are required to control radiation exposure are reflected in insurance, and thereby in the cost of nuclear energy. The funds which are allocated to industrial safety in atomic energy installations must be allocated in such a way that the relationship of radiation exposure to the broad picture of industrial hazards is recognized.

In view of the involvement of the general public in radiation protection problems and to the large number of workers now involved in the industry, this problem has and will continue to receive a good deal of attention from Federal and State governments. The semi-voluntary character of the practices established by the NCRP can no longer be maintained; the authority of government is being exercised in this field. The problem from the point of view of the legislator is to provide a force of law without impeding the influence of new information from research on radiation protection practices.

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SYMPOSIUM ON THE EFFECT OF THE NCRP RECOMMENDATIONS ON NATIONAL LIFE

I. EFFECTS ON LABOR*

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I HAVE been asked to comment upon the effects of the NCRP recommendations on national life, and, more specifically, to present the viewpoint of Labor as to these effects.

It is well known that Labor is deeply concerned with the safety of industrial employees. This concern is manifested in many ways, most of which are familiar to you, and include such safety activities as the American Standards Association and others.

With respect to the subject of radiation and radiation safety, however, I should like to develop briefly, the organization within the AFL-CIO which concerns itself with this problem only. Under the general heading of The Committee on Economic Policy, which is composed of international presidents, is the "Staff Subcommittee on Atomic Energy and Natural Resources". This staff subcommittee is composed of delegates from all International Unions, affiliated with the AFL-CIO, who have an interest in atomic energy. This will include delegates from the Metal Trades Department (whose affiliation alone accounts for two-thirds of all organized radiation workers in AEC facilities and many thousand additional workers in plants operated by licensees and contractors of the AEC) as well as the Boilermakers International, the Electrical Workers, the Oil, Chemical and Atomic Workers, Pipe Fitters, and many others. As usual, this staff subcommittee has additional subcommittees. As an example, one such subcommittee, of which I am Chairman, is specifically titled "Subcommittee on Radiation" whose responsibility is to follow activities in this area.

It is the purpose of these various committees to discuss and correlate activities in their field, as well as to advise the large Policy Board on matters falling within its purview. This has proved to be quite an effective method of organization, and the committees have labored long and hard, particularly within the past few years.

We have felt strongly that the time has come (indeed, it arrived some time ago) when broad, intelligent discussion and action can begin on the establishment and administration of radiation protection standards. There are important decisions, economic, social and political in nature, which must be made in the formation of a national policy which will be acceptable not only to Labor, but to the public as a whole.

Fundamentally necessary in the formation of any such national radiation policy, is the criteria which will be used in order to establish such a national policy on an acceptable basis. These criteria must be established by an absolutely impartial group. This is to say, a group which will favor neither Labor, nor Industry, nor Government, nor Science, but will serve all equally and indiscriminantly. This must likewise be composed of persons enjoying the highest standard of respect as to their technical abilities, as well as to their humanism. It is our opinion, by and large, that the National Committee on Radiation Protection and Measurement meets these qualifications.

The criteria used in the establishment of safety standards must be based upon the reduction to the minimum of the harmful effects of radiation exposure, and of course, in the event of a

* Presented at the Fifth Annual Meeting of the Health Physics Society, Boston, Mass., 30 June 1960.

question, such questions must be received in favor of public health and safety. We feel that this is the direction in which the NCRP is proceeding. The time for decision on standards comes when a given level of exposure is placed in balance against the benefits sought, stemming from nuclear development. This decision must, of itself, be a moral and social one. This follows because we are dealing with benefits which are expected to accrue to human beings, and risks to which they are exposed in the process. Such effects can never be resolved solely on scientific bases.

Dr. ABEL WOLMAN stated: "Criteria must rest upon public health protection and not cost. No one, of course, should advocate excessive and unnecessary restraints. These restraints, logically suggested, however, within the framework of current scientific understanding, should not be resisted solely because resulting costs may threaten to throttle application . . . ." He further stated that "The guiding principle must be maximum protection of people, not because of sentiment, but because society demands it. An agreed acceptance of a number of consequent disabilities is not an appealing basis for the development, say, of nuclear power. Industry will do better than rest upon such an affront to man".

The Federal Radiation Council, in its radiation protection guidance for Federal agencies, states that "Fundamentally, setting basic radiation protection standards involves passing judgment on the extent of the possible health hazard society is willing to accept in order to realize the known benefits of radiation". This brings us logically to mention the concept that organized labor cannot, as a public interest group and making an effort to represent correctly the opinions of its members, agree that any stipulated radiation dose is an "acceptable risk". This is not to say that there are no members of organized labor who will not accept a certain risk in its day-to-day occupational activities. Obviously this is not so. What we mean to say is, that we cannot willingly accept the concept that a stipulated radiation exposure is acceptable in return for *X* numbers of dollars. Putting it another way, we feel that it is inevitable that a worker trained in the radiation industry and enjoying a low occupational exposure record will attempt to utilize this low record as a merchantable item, and this to the detriment of others who may be less fortunate.

Another problem we see in conjunction with this, is that there appears to be no threshold dose, and it becomes commonplace when a given maximum is presented, as in the case of the NCRP recommendations, to utilize these maxima as a "do not exceed" figure, and consider that any dosage up to the limits thus recommended is "acceptable", rather than utilizing these maxima and working downward from them in such a way as to drive these maximums lower and lower as rapidly as technology and the ability to do so permits. There is no question that there exists today some industrial concerns who consistently and intentionally permit the exposure of their employees to the maximum permissible dose. Consider, for instance, the testimony before the Joint Committee of a processor of isotopes who contended that any lowering of the NCRP recommendations and concommitantly the IO CFR Part 20 of the AEC would drive him out of business; that his practice was continually to expose his employees to the absolute maximum permitted by these recommendations and regulations. There are probably a large number of persons active in the field of health physics, as well as in the AFL-CIO, who have a strong feeling that, if necessary, employers of this type would have no hesitation in permitting the exposure of employees to doses greater than the MPD, if the occasion warranted.

This testimony and others have led the AFL-CIO generally to adopt the position stated above—that standards should be continually driven downward, and not to accept passively the present levels of exposure. For this reason, we in Labor have been delighted to observe the continuing downward revision of the NCRP permissible levels. For this reason, we feel that the recommendations of the NCRP, both in themselves and as codified by the AEC, are not simply a cold set of numbers, but are, in fact, a whole philosophy of a way of living and working.

There is a continuing assault against this philosophy in the form of a tendency to relate radiation hazards to other industrial hazards.

On the surface, this is probably as it should be. Yet, it cannot be done so entirely, because radiation is a unique hazard. This is complicated by a certain complacency which has developed because of the excellent safety record thus far in the atomic energy field. The AEC, for instance, and its contractors have enjoyed remarkably low accident rates as compared to those achieved in other industries. The intrinsic nature of work in the field of atomic energy is in itself hazardous and relates in kind and degree of hazard to those industries (except for radiation) to that of refining, smelting and chemical processing. Radiation adds to the commonplace hazards of these industries rather exotic new elements, and that the accident rates, both as to severity and frequency, are so low in an industry which began slightly more than fifteen years ago is, indeed, remarkable.

I think the Health Physics Society probably feels, along with me, that this is not due entirely to the lack of occasion for such accidents, or to the lack of circumstances under which accidents could readily have happened, but is due, rather, to the all-out drive to keep such accidents at their irreducible minimum. Thus, the complacency is not justified, particularly in the light of the tremendous amount of effort and money expended toward keeping the industry safe.

It has been argued, and we tend to agree, that the economic burden of lowering exposure limits is greater for smaller companies where radiation protection costs account for a relatively higher percentage of operating costs than would be the case in a larger company. We also suspect that for this reason these smaller companies will tend toward the higher exposure rates.

Radiation exposure costs are contingent on many things, and thus it is not safe to generalize, as in the case of a power reactor operation, for instance, one estimate showed that if current limits were increased ten times there would be a 5 per cent decrease in operating radiation protection costs. If current limits were lowered ten times, there would be a 50 per cent increase in these costs. It seems, therefore, that design is the answer here. In the above case, lowering limits would increase costs by making alterations in programs necessary to compensate for reactor design either in leakage rates or handling, or whatever. This is shown by the relatively modest decrease in costs occasioned by raising limits. One aspect of the effects of the NCRP recommendations which vitally concerns Labor is that which deals with the new law (Public Law 86-373) which invites participation by the states in the regulation of nuclear materials and sources. It is apparent that the states will be obliged to develop a codification of radiation safety regulations which will be "compatible" with those which are promulgated by the Atomic Energy Commission in their IO CFR Part 20. We wish to point out here that there is a real danger that there will be no organized effort to bring the word "compatible" to mean that the new regulations will be uniform to the extent that is required for effective control of the hazards from one state to the other. We would have preferred that the word "compatible" be exchanged to read "identical".

The "Radiation Protection Guide" issued by the Federal Radiation Council itself states in its recommendations: Part 7.10 "There can be different Radiation Protection Guides with different numerical values depending on the circumstances. The guides, herein, recommended are appropriate for normal peacetime operations." Thus, there is permitted, within Government itself, a latitude whose bounds are not yet determined. Then, if this usual latitude is not yet sufficient, Part 7 of the "Guide" states: "The Federal Agencies (should) apply these Radiation Protection Guides with judgement and discretion, to assure that reasonable probability is achieved in the attainment of the desired goal of protecting man from the undesirable effects of radiation. The Guides may be exceeded only after the Federal Agency having jurisdiction over the matter has carefully considered the reason for doing so in the light of the recommendations of this staff report." To use Dr. Dunham's celebrated analogy about speed limits, I submit that if speed limits were written as intentionally devious as the above, there would never be a single speeding arrest made. The whole of the "Radiation Guide" is shot through with similar expressions, and I hope I have made the point that we cannot expect uniformity in state standards when it is apparently impossible for the Federal Government

to require that its agencies themselves adhere to a firm set of numbers, practices and procedures, despite the fact that one of its agencies has chartered the way for nearly fifteen years with an enviable safety record.

Based upon these and other actions, it is the opinion of Labor that the FRC would benefit substantially by the addition of public members, and we have so testified this thought to be necessary, since the FRC is not subject to the Administration Procedures Act which would require public hearings on actions and proposed regulations of the Council. It appears, however, that there is an effort to keep this a governmental club despite the intent and language of the Public Law which establishes it. We have, therefore, renewed our request to the JCAE to establish a statutory advisory Committee to which public members (meaning Labor and Industry) will be permitted to voice their opinions as to the problems which they at first hand have encountered.

I do not wish to be unduly critical nor unjust in my remarks, but it is felt that these "Guides" are not a product worthy of such a highly constituted board, and I predict that in the indefinite light of the "Guides" it will be an impossibility to bring the states into line behind such a word as "compatible" in order to insure uniformity.

With one further reference to the deficiencies of the "Guide" I shall leave them. That reference is to the complete omission from the "Guide" to any mention whatever of records of radiation exposure to personnel. There is no requirement as to these records, how they are to be kept, where they are to be kept, or even a requirement that the agencies note such exposures as are permitted. It is also difficult to conceive of a single instance where uniformity is more to be desired than that records be maintained and in what fashion. The necessity for this is apparent to everyone in this room, yet the fundamental release by the Council which is charged with the responsibility of advising the President on matters affecting radiation, and which brought forth its basic guidance for governmental agencies, has omitted any reference whatever to the need for such bookkeeping by those agencies.

I wish to make one further point with respect to the states and their activities in regulating radiation facilities. The States have had for nearly 40 years the problem of radiation. How well they have handled it is questionable, and for part of the answer I refer directly to Dr. S. H. VALE's testimony (3 June, 1960) to the subcommittee of the JCAE. Dr. VALE represented the American Dental Association and estimated that 83,000 dentists operated approximately 95,000 dental X-ray machines. In his testimony he was frank to state that there was as yet a significant fraction of the machines in use today which were either uncollimated or unfiltered or both. From November 1957 to March of 1958, a survey of 122 machines in 113 offices in the Chicago area showed that only 22.6 per cent were collimated and only 16.8 per cent were filtered. From 2 to 3 years later, a nationwide and current survey by ten colleges of 2000 offices and 1649 machines showed that 69 per cent were collimated and 56.3 per cent were filtered. Putting reverse spin on this we find that nearly one-third of all the machines in the United States are uncollimated and nearly half are unfiltered. And this is an area where States, which are now professing great interest in the regulation of atomic energy and the more glamorous types of radiation, could have functioned for nearly a half-century.

Since I have departed somewhat from my assigned topic, I should like, with your permission, to digress a bit further to mention a topic which should be of interest to you. We feel that while there are undoubtedly two levels of radiation control people (the professional and the journeyman) we do not believe that placing them under various titles such as Radiation Monitors, Rad-Safe Man, Radiation Safety Engineer, Radiation Security Officer, Junior Health Physicist, and nearly as many other titles as there are facilities, is a phase of the art or profession that will insure confidence of either industry, labor, or the general public, because of the resultant confusion. We are concerned with the educational and experience background of the people who are or will be developing procedures, analyzing hazards, acting as consultants for formal industries or businesses, and other top-level health-physics positions;

and are delighted to see that efforts are being made for the qualification of such professional personnel.

The second level, or the people who make surveys, police the procedures, and generally control the hazard at first hand are probably not the same in training or responsibility. But they, too, must be trained in their jobs to the extent of assuring confidence, skill, techniques, experience and a good relationship with production and supervisory personnel. We feel that these people need not necessarilly have a B.S. or Ph.D. degree, but it should be required that they have at least four years on-the-job training and experience as a formal apprenticeship, which will ultimately lead to journeyman status.

I, therefore, suggest to you the establishment of a committee composed of equal numbers of members of the Health Physics Society and the AFL-CIO to explore this latter possibility and attempt to arrange a closer relationship which will round out the health and safety personnel protecting the people employed in the atomic industry. The problem of qualification and training of this "journeyman" level of safety personnel ranges from those employed on construction, and in industry, as well as by AEC facilities and licensees. This problem is the same as that encountered in the qualification of the professional, and such a committee as I have suggested may well be a step toward its solution.

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SYMPOSIUM ON THE EFFECT OF THE NCRP RECOMMENDATIONS ON NATIONAL LIFE

II. LEGISLATIVE ASPECTS OF RADIATION STANDARDS: ROLE OF NCRP AND GOVERNMENT

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Chairman of the Special Radiation Subcommittee of the Joint Committee on Atomic Energy

THE recommendations of the National Committee on Radiation Protection were a subject of considerable discussion at the hearings on Radiation Protection Standards recently completed by the Special Radiation Subcommittee of the Joint Congressional Committee on Atomic Energy. Several distinguished members of the Health Physics Society participated in that discussion in oral testimony at the hearings and many others made valuable contributions to the volume of Selected Materials printed in advance of the hearings.

Those hearings, I may note, reflected once again the Joint Committee's continuing interest in the subject of radiation hazards posed by the expansion of nuclear technology and the application of that technology to peaceful as well as military uses. The Joint Committee has endeavored to ensure that the full benefits of our nuclear resources will be enjoyed by the American people, but at the same time, we have made every effort to understand and provide safeguards against the hazards posed by the development of these resources.

The effect of the NCRP recommendations on national life is not easily determined. One must consider not only the uses to which the specific NCRP numbers are put, but also the influence which the philosophy reflected in those numbers has on various governmental bodies and other institutions as well as professional groups considering problems of exposure to man-made ionizing radiation.

The use of NCRP numbers is more easily ascertained than is the influence of NCRP philosophy, though in my judgment the latter is of much greater significance.

The NCRP philosophy, or attitude and treatment, in dealing with broad questions of radiation protection in general represents the collective judgments of competent scientists who have devoted considerable study to this field. In that sense, it is representative in nature. At the same time, however, the expressions of this body carry great weight with the scientific community and tend to set the tone and furnish direction to scientific investigations in this field. In this sense, the NCRP is a leadership group. It focuses attention on particular problems; it offers suggestions as to the limits of safety in various radiation exposure situations; it indicates additional subjects which need further exploration.

How the NCRP acquired this role in our society is a worthwhile question to consider. In the absence of Government action this voluntary technical group filled the vacuum.

Thirty years ago, when the NCRP was established, no one could foresee the widespread need for radiation protection standards that confronts us today. The original NCRP effort was directed toward the development of standards for the safe handling of radiation instruments used by physicians and physicists. It was essentially a self-policing effort aimed at preventing demonstrable injury in the use of X-ray procedures for clearly indicated benefits.

Today, with our present knowledge of radiation effects, we may consider the early NCRP standards as rather crude, but the continued

* Presented by CAREY BREWER at the Fifth Annual Meeting of the Health Physics Society, Boston, Mass., 30 June 1960.

study of radiation protection techniques and the development and refinement of concepts has produced a body of knowledge and experience of great import to all users of radiation and to the handling of radiation sources.

A growing recognition that not just the users, but also persons in areas adjacent to radiation, installations were being exposed to radiation, opened up a broad new area of study and led to the development of new concepts of exposure limits for the population at large as opposed to those occupationally exposed to radiation.

The fact that the effects of radiation exposure might not be immediately evident, but would appear only after a long latent period, has generated increased concern with respect to recipients of radiation treatments as well as with the increasing exposure of the general population from sources other than natural background and the original uses of X-ray.

The extension of NCRP efforts to these comparatively new areas seems only a natural exercise of its functions since it has been the only national scientific organization of its kind in the field. Nevertheless, it is an unofficial one, and its recommendations are not automatically incorporated into law or into rules, regulations or practices based on them by the various levels of government.

In recent years, the Joint Committee on Atomic Energy has become increasingly aware of the role being performed by the NCRP. This has resulted mainly from our studies and investigations of the effects of fallout from nuclear weapons testing, radioactive waste disposal, employee hazards and workmen's compensation, and Federal–State relations in the atomic energy field, as well as from our recent examination of the basis and use of radiation protection standards.

The increasingly significant role of NCRP in setting the pattern and actually supplying the numbers on which major governmental decisions are made in the field of radiation protection, is in my opinion cause for an examination of the functions performed by government in this field. The basic issue in question is whether government is properly discharging its responsibility or whether it is relying too heavily on the guidance of a quasi-private group.

This is not a question of the inherent value of NCRP recommendations or the validity of the scientific findings of that body. It is purely a question of functions that must be performed by Government—at all levels—when rules are established for the protection as well as the regulation of society at large or any major segments of our society. In the case of exposure to man-made sources of ionizing radiation, the setting of standards can no longer be considered solely a self-policing function to be performed by the users of radiation sources.

There is a significant continuing role to be performed by groups such as the NCRP and its international counterpart, the International Commission on Radiological Protection (ICRP). There will always be a need for independent collection and examination of scientific data. Moreover, independent analysis of problems and appraisals of alternative courses of action, as well as criticism of public policy, will always contribute heavily to the advancement of protection in this field and will tend to ensure responsible governmental action. But the final judgments going into the makeup and application of standards as public policy must be made by government.

The governmental function to be performed is not limited to the imposition of orderly procedures and providing legal remedy or administrative recourse for parties adversely affected by the exercise of public policy. These are indispensable attributes of representative government and are highly significant in the field of radiation protection. But a larger aspect of the governmental function is the making of value judgments, the weighing of economic, political and social considerations, which is a function no private group such as NCRP should be responsible for performing. I hasten to add that, to my knowledge, NCRP has never intended that its recommendations be used by government agencies without further consideration of economic and social factors.

Within this general framework of discussion it is interesting and, I believe, instructive to consider the role of the Federal Radiation Council, which is the repository of radiation protection policy at the Federal level.

Last year the FRC was first established by

Presidential order and later given a statutory basis by Public Law 86–373. That law provided for a council of high Federal offiicals* who should "advise the President with respect to radiation matters, directly or indirectly affecting health, including guidance for all Federal agencies in the formulation of radiation standards and in the establishment and execution of programs of cooperation with States".

It is significant that the statute specifically provides that the Council shall consult qualified scientists and experts, including the NCRP Chairman.

The text and the legislative history of the statute indicate that the FRC was intended to serve a function quite different from that carried out by the NCRP. It was not created for the purpose of duplicating the basic scientific investigations and collection of data which characterize much of the effort of the NCRP and other independent bodies. Its intended function was that of bringing available scientific data to bear on policy decisions affecting the use of radiation protection standards by agencies of the Federal Government.

In other words, for the first time a Federal body was charged with the responsibility of weighing the social and economic gains—not to mention considerations of national defense—against the biological risks of man-made exposures to ionizing radiation. This involves the weighing of gains against the risks to society as a whole or to particular groups within our society.

Those of us who had entertained hopes that the FRC would actually carry out this highly important function have been somewhat disappointed by the operation of the FRC to date. There is little evidence that the FRC has taken into consideration social and economic factors in setting radiation standards, especially for the population as a whole. The hearings brought out that although the Council did have the advice of various technical consultants, there was no consultation with social scientists and groups interested in the broader problems.

The one memorandum issued to date by the FRC changes the NCRP terminology of "maximum permissible exposure" and "maximum permissible concentrations" to the new terms of "Radiation Protection Guide" and "Radioactivity Concentration Guides", which are to be used by Federal Agencies. The specific numbers to be used in Agency regulations setting radiation standards are essentially those currently recommended by the NCRP.

Whatever value this FRC memorandum might have achieved in providing direction to Federal Agencies in the application of radiation standards was partially nullified by a final provision which states:

"It is recommended that:

7. The Federal agencies apply these Radiation Protection Guides with judgment and discretion, to assure that reasonable probability is achieved in the attainment of the desired goal of protecting man from the undesirable effects of radiation. The Guides may be exceeded only after the Federal agency having jurisdiction over the matter has carefully considered the reason for doing so in light of the recommendations in this paper.

"The Radiation Protection Guides provide a general framework for the radiation protection requirements. It is expected that each Federal agency, by virtue of its immediate knowledge of its operating problems, will use these Guides as a basis upon which to develop detailed standards tailored to meet its particular requirements. The Council will follow the activities of the Federal agencies in this area and will promote the necessary coordination to achieve an effective Federal program."

Aside from the curious fact that the FRC adopted the NCRP numbers and then left the door open for noncompliance by Federal Agencies, several basic questions have been posed as to the role of the FRC. Some of these are:

(1) Where does final responsibility reside for the imposition of standards embodied in Agency regulations affecting the economic and social welfare of the public?

(2) Must the standards applied by individual

* The Secretary of Health, Education and Welfare; the Chairman of the Atomic Energy Commission; the Secretary of Defense; the Secretary of Commerce; the Secretary of Labor; and such other members as the President may appoint.

Federal Agencies be justified in accordance with established administrative procedures within the Agencies, or does the FRC memorandum, which was issued as a Presidential directive, remove conventional requirements of this type?

(3) To what extent has the Federal Radiation Council relieved the individual agencies of responsibility in making value judgments of overall economic and social import?

(4) If a number of individual agencies are authorized to set standards of radiation exposure within the confines of permissible limits or "Radioactivity Concentration Guides", for general population exposure, who allocates quotas to ensure that the total does not exceed the established limits? Is each agency free to pre-empt the full limits of exposure in the entire pool in pursuing its own objectives? Should not each significant additional contribution be reviewed by the FRC in order to consider each incremental contribution in relation to the whole?

Some of these questions were discussed at our recent hearings, but thus far the answers have not been found. Of particular concern to members of the Committee and witnesses alike was the problem of providing recourse for parties adversely affected by FRC recommendations and the general question of obtaining representation for particular groups having a direct economic or other interest in the development and application of radiation standards by the Federal Government. As a result of our question on this point, Chairman FLEMMING of the FRC indicated that they would devise some means of dealing with this problem.

Another question of direct concern to the Joint Committee on Atomic Energy is the extent to which the FRC is accountable to the Congress, which in our representative system has special obligations in overseeing the operations of the Executive Branch.

The FRC, I may note, is in the anomalous position of being advisory to the President while at the same time serving as the President's agent, and, according to the testimony of FRC Chairman FLEMMING, constituting a review body for ascertaining compliance in this field on the part of other Federal agencies. As a Presidential advisory body, the FRC has on one occasion claimed executive privilege in declining to furnish Congress with information concerning its operations. Thus, even Congressional review of FRC activities will be limited until corrective action is taken to clarify the position of the FRC.

Interestingly enough, these multiple duties are performed by the FRC with the assistance of only one full-time professional staff member.

It should be noted that questions now posed with respect to the proper role of the FRC also carry implications concerning the future role of the NCRP. For example, should the NCRP be made a formal advisory group to the FRC? Should it become a private non-profit corporation? Is there a conflict of interest in the case of Federal employees serving as members of the NCRP? What are the proper limits of Federal support of NCRP activities which could conceivably produce findings in conflict with those of the FRC?

I have no information as to the plans which the NCRP may have concerning its future role, but in my view it presently performs an extremely valuable sevice as an independent body free of official policy restrictions. The impact which its recommendations have had on national life is attributable, not to any official status on the part of NCRP, but to general acceptance by government and private groups that have chosen to follow those recommendations. I personally hope that nothing will be done to disrupt the fine work of the NCRP and that means will be found to permit the continued participation of Federal employees and officials in the work of this group.

In my judgment, the pressing need in the field of radiation protection is to establish appropriate Federal machinery for broad policy decisions that must be made. If the Federal Radiation Council cannot be modified to perform this function, or if it is unwilling to do so, then the remedy may lie in replacing the FRC with a different type of Federal agency.

It has been suggested that an independent regulatory agency, with appropriate hearing procedures and rule-making powers, could more properly discharge the Federal Government's obligations in this field. It is possible that the Joint Committee on Atomic Energy will want to consider various alternatives in the next session of Congress.

Whatever determination is made in this respect, it is clear to me that the present FRC has not comprehended the broad policy implications of the task assigned to it by law. If the meaning is unclear or the authority is uncertain, Congress will be compelled to come forward with corrective legislation.

I may add that improved Federal organization is not the sole answer to the various problems and issues of radiation exposure which have been accentuated by our burgeoning nuclear technology. The development and refinement of new concepts in radiation protection and the accumulation of new scientific data are functions residing with the scientific community at large, and in the long run these functions will influence decisively the functions performed by government at all levels.

It has been said that science and technology have advanced so rapidly in recent years that society has been unable to keep pace; that a runaway technology now threatens to overwhelm government and society at large. Whether this is true or not, the statement carries an implication that government ought to control or direct the course of scientific advances. The fact of the matter is that government cannot control the advancement of science, although it can and does at times influence its application. In the main, science moves ahead, sometimes helped and sometimes hindered by government, but it keeps on moving.

The governmental task is to encourage the application of science to the larger benefit of society and, if possible, to reduce the harmful or injurious impact which advancing technology might have on society.

I believe this is precisely the task we face today in attempting to adjust and balance the benefits and risks of ionizing radiation in the field of nuclear technology.

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SYMPOSIUM ON THE EFFECT OF THE NCRP RECOMMENDATIONS ON NATIONAL LIFE

III. EFFECTS ON MEDICINE*

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THE general effect of the NCRP recommendations on medicine and its progress has been wholesome. Far too many of the physicians using ionizing radiation in the past had not had adequate training in physics and radiation protection and without the help of the NCRP recommendations might well have done harm to their patients, their employees and themselves. One only needs to remember the hundreds of names on the monument to the early martyrs to radiation in St. George's Hospital at Hamburg, Germany, to realize the great value of the NCRP recommendations. Fortunately, however, these recommendations are flexible and have been so framed as not to infringe on the judgment of the physician in arriving at the diagnosis of the lesion from which his patient is suffering nor to hamper him in the use of ionizing radiation for purposes of treatment.

To go back only a generation ago, I am sure that more pathologists in training died from occupationally acquired pulmonary tuberculosis than radiologists in training were killed or injured by ionizing radiation. Had not X-rays been freely used to detect early pulmonary tuberculosis and permit treatment in a then curable stage, far more pathologists would have died of tuberculosis than would have been saved from the possible hazard of diagnostic use of X-rays by restriction of X-ray use.

I would like briefly here to pick up a point that Mr. BREWER made. While in general I have found myself in agreement with what he said, there must be discretion allowed to Federal agencies as well as to others in times of emergencies. Let us suppose, for example, that at the time of the bombing of Hiroshima and Nagasaki, it had been decided from calculations that entry into the ruins of those cities and determination of what had happened to the survivors were too dangerous and the personnel of the Manhattan District, of the Army and the Navy had been prevented from going in. We would have been infinitely more ignorant about the effects of radiation than we are at the present time. Let us take another example. In Tokyo I saw an ill radiologist. He was ill because he had had the task of fluoroscoping a great number of the prisoners of war who had been in the hands of the Japanese to determine those suffering from tuberculosis. Because he was the best skilled American radiologist in that area at the time, he knowingly exceeded the NCRP precautions under pressure of the work that he had to do, and he saved a great many lives by so doing. If his commanding officer had had to tell him, "You don't do more than two hours of fluoroscoping today," that would have been tragic. Rules in general are guides, and there are times when one has to depart from them.

A few further points as to the effect of the NCRP on medical affairs: First, a great many radiologists are alive today who would not be had they not had the guidelines of the NCRP. These guidelines have changed downward a number of times and have changed for four reasons:

(1) We have learned more.

(2) We have better instruments.

(3) The character of radiation used has become much more penetrating and, hence, much more of a risk to the internal tissues.

(4) Many more people were coming in contact with radiation, and one had to begin to weigh genetic factors. The geneticist considers

* Presented at the Fifth Annual Meeting of the Health Physics Society, Boston, Mass., 30 June 1960.

that 100 r delivered to the gonads of any one person is the same as one-thousandth of a roentgen delivered to 100,000 people. As the potentiality of exposure reached the population as a whole through radioactive fallout, and became greater through wider employment in industries utilizing radiation, it became necessary to lower the exposures because of genetic considerations.

Now, again, permit me to take issue briefly with Mr. McGowan. He suggested that we should push down radiation levels, lower and lower and lower still. This is good perhaps from one standpoint, but we are fast approaching the point, if we push them much lower, instrumentation will not only be difficult but the results meaningless because of the approximation of natural background, with its variations, which we cannot control. We must look at the paradise of no radiation exposure as utterly unachievable in this life.

One has to weigh what has been the experience of mankind, of animals, of plants, over the long period of their past existence and how this relates to the levels that are proposed.

It is possible to get ourselves in serious trouble by being too afraid of radiation. There are more people today who worry about having necessary diagnostic X-rays than those who are too little concerned over excess radiation. By that I do not mean that any unnecessary risk should be willingly suffered. However, I think that we must keep in mind that it is utterly impossible to escape radiation unless one is content to stay in a submarine at the bottom of an ocean or to live in the bottom of a mine in rocks free from any trace of radioactivity. We must learn to live with radiation. We learn to live with a number of things, though it is a hard process.

What happens when a superficially desirable but intrinsically silly law to protect health is enforced? This is the law that was passed with the best intention in the world, a law which says that no substance which has caused cancer in animal or man shall be added to food. Perfectly reasonable at first glance, but this law does not consider the concentration of the carcinogenic material. As a result, a pronouncement from the Department of Health, Education and Welfare ruined the livelihood of hundreds of farmers here in Massachusetts and elsewhere, and you, as taxpayers to the Federal Government, are paying damages to them, because traces of a weed killer led the Department to make this pronouncement. One would have to eat many grams of this material daily to get the same dose that produced cancer in relatively few mice. In other words, if a good rule is pushed too far, it becomes a bad one. I think one of the fine things about the NCRP over the years has been that it has had support from among the radiologists who know the good that they can do with radiation and try, as was so well brought out by Mr. McGowan and Mr. Brewer, to keep a proper balance between the value received from radiation and the harm that might be done.

There is still another impact of the NCRP and its rules: the utilization of radioactive isotopes for research purposes. The bulk of work in the field of experimentation with radiation can be done on plants and animals, but a cornstalk or a mouse are not men, and we have to have some knowledge that comes from man if we are to do good to men. For example, a quick determination of the volume of his blood following a severe accident may determine the need or lack of need for transfusion. New procedures can be worked out initially in experimental animals but finally they have to be worked out in man. The NCRP guides have been of tremendous help in determining the permissible levels of isotopes to use in human experimentation, experimentation that is essential for the advancement of human welfare.

The NCRP rules have been so drawn that this important field has on the one hand been adequately controlled, on the other hand, not unduly hampered. If one pushes permissible doses too low, medical practice will be hampered because there are some things that can be done only with isotopes. I thought my laboratory was highly competent some years ago when, as a result of an analytical procedure that took about 3 days, we could detect one gamma of phosphorus. Just think how simple, easy and quick it is to determine the equivalent of one gamma of phosphorus-32 through its radioactivity.

Our complete understanding of the utilization of food by man was revolutionized when radiocarbon became available, because up until that

time, some of the steps in the processes of utilization were happening so fast that no chemical technique could hope to follow them. By the time the analyses would have been done, the compound would have been gone, and it was only through radioactive isotopes that intermediary metabolism could be followed.

In pushing permissible levels of radiation down, therefore, weigh carefully lest essential advances be hampered.

From the positive standpoint, let us say this for restrictive rules. There are recognizable occupational hazards of radiologists. Until 1955, the American radiologists appear to have had a somewhat shorter life than other practitioners of medicine. The statistics as such are very hard to interpret and one would hesitate to say how much shorter. Radiologists have had seven to ten times as much leukemia as other practitioners of medicine. Much of this unfavorable experience goes back to the early workers in radiation, before the NCRP got started.

The use of the present permissible levels cannot be demonstrated to have produced any harmful effects. In the period from 1955 to the present, the life span of radiologists has appreciably lengthened because the weighting due to the exposure to the pioneer workers is no longer present.

Still another problem is whether to consider medical and diagnostic X-rays, medical and diagnostic use of radium or other radioactive substances as additive to the occupational exposures. I know Mr. WILLIAMS is going to say something about this very complicated matter later. Let us say, for example, that a man has an ulcer of his stomach, and he also happens to work in radiation. (There are a fair number of us who are inclined to get ulcers.) Shall his doctor have to make a choice between giving his patient a necessary X-ray and having him laid off the job for a period? I am quite sure that ulcer patients would not be hired very long by industry if this were the case. Every rule must be kept sufficiently flexible so that paramount needs of the individual can transcend that particular rule if his welfare demands it. This does not mean that there should be carelessness or reckless use of radiation, and sometimes because of lack of supervision over the patient or equipment this has happened. I remember the shock that I received in looking over a group of measurements that had been made in tuberculosis detection units. One set was almost incredibly high. Investigation showed the technician in charge of the apparatus had decided he got better results if he left out the filter and the collimating device and consequently was shooting the full beam squarely at the patient without proper protection. There are always similar errors that one has to guard against.

One can see very useful things happen when a demand is created. One of the effects of the NCRP regulation has been to force radiologists to maintain instruments that would give equivalent or better results from less exposure. It is really impressive to look back at the types of X-ray apparatus that were in use in 1930 and contrast the pictures taken and the exposures given with the results and exposures at the present time. Much of this progress has been stimulated by the NCRP and its rules.

Don't mistake me—I am not speaking against rules; they are absolutely essential. No one believes more firmly than I in keeping radiation at a minimal, but let us not make the rules so restrictive that there is not a possibility of the essential flexibility that will permit the welfare of the individual; the welfare of society as a whole, must be kept in mind also. Don't make the rule the absolute but rather make it the relative guide.

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SYMPOSIUM ON THE EFFECT OF THE NCRP RECOMMENDATIONS ON NATIONAL LIFE

IV. EFFECT ON INSURANCE*

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IT IS a well-known fact that the recommendations of the NCRP have long been the basis of various kinds of standards in the radiation field. The history of the development of these standards has been told many times. While the Committee originally started for the purpose of providing a basis for protection in the medical field, its activities have been expanded over the years, first, to include the problems of industry and, more recently, to include exposures to the general population from such sources as fallout. One of the outstanding results of this was brought about by the fact that as the numbers of exposed people increased significantly, genetic effects became more important and finally the controlling factor. The result of this trend must, of necessity, be in the direction of lower and lower values for recommended maximum permissible exposures. The original ICRP value established in 1934 was 1 r per week. This number was followed until approximately 1950. NCRP at the same time (1934) adopted half the ICRP value or 0.5 r per week. In 1949 the NCRP again lowered its recommended value for occupational exposure to 0.3 r per week. This number was applied until the latest recommendations of the NCRP dated January 1957, lowering the value to 0.1 r per week. At the same time, the concept of the application of differing values, depending upon the age of the worker, the kind of radiation to which he was exposed, and portions of the body involved was introduced. These changes, while based upon the concepts of genetic damage, actually resulted in a kind of individual standard. The most recent recommendations also include levels for internal emitters in air and water as well as for body burden. In addition to the industrial kinds of exposures, NCRP in 1952 began to give consideration to limits of dose to the general population as a result of man-made radiation exposures. The basis of these numbers is again primarily the genetic effect resulting from exposing very large segments of the population to low levels of radiation. Thus, we have seen during the past few years the impact of the geneticist on our radiation levels, both in the industrial field as well as on application to the general public.

In order to examine the effects of these recommendations on the national life, we must bear in mind that there are three kinds of exposures with which we must be concerned. In addition to the industrial applications with which we are all so familiar and the medical applications which provide the population with far more radiation exposures than any other, we must also consider the small but widespread amount of exposure resulting from fallout.

The actual effect of the recommendations will depend upon the way in which they are applied. The recommendations, as they are published, do not carry any force of law, but become effective only when some regulatory body utilizes them. It is significant that these values are generally accepted as a basis for regulations of all kinds. The Atomic Energy Commission utilizes these numbers in the promulgation of 10 CFR 20. The various States have applied NCRP recommendations in their codes and regulations. In addition to this, groups engaged in standards work such as the American Standards Association, the American Industrial Hygiene Association, the American Conference of Governmental Industrial Hygienists base their radiation levels on NCRP recommendations. It is thus obvious that the NCRP is widely accepted as the one significant group

* Presented at the Fifth Annual Meeting of the Health Physics Society, Boston, Mass., 30 June 1960.

in this country which carefully examines all the available data on radiation damage and provides the best possible information to be utilized as a basis for radiation protection work. The quality of personnel and the thoroughness of their work is the real basis for this acceptance.

The Federal Radiation Council, which was established in 1959, has issued its first document which comprises a series of seven recommendations to be followed by the various federal agencies in their radiation work. This first release does two basic things. First, it establishes on a Federal level the term "Radiation Protection Guide", which is defined as "the radiation dose which should not be exceeded without careful consideration of the reasons for doing so; every effort should be made to encourage the maintenance of radiation doses as far below this guide as practicable". A series of numbers—Radiation Protection Guides—which are essentially the values from the most recent NCRP recommendations, are set forth as the radiation protection guides to be used by all Federal agencies. The Federal Radiation Council now has a task force working on the problem of internal emitters, and it is assumed that the next release of the Federal Radiation Council will be a series of recommendations relating to these substances.

In order to keep its recommendations as flexible as possible, Recommendation No. 7 provides that a Federal agency shall use these guides with "judgment and discretion" and then goes on to state "the guides may be exceeded only after the Federal agency having jurisdiction over the matter has carefully considered the reason for doing so in the light of the recommendations in this paper". The implications of this section are obvious.

One cannot help but wonder, after reading the first releases of the Council, what the ultimate fate of the NCRP is going to be. It is, therefore, timely that we review the impact of its latest recommendations on atomic energy operations and on the American people.

IMPACT ON INSURANCE

Many people look on insurance solely as a device for financial protection through the spread of risk. Under the modern concept of insurance operations, it is actually far more than this. Insurance companies are concerned not only with the payment of loss but also of the control of loss through the prevention of accidents and reducing the spread of damage after accidents have occurred. The kinds of loss with which we are concerned fall into two major categories. The one which has the greatest potential is related to risk of injury or death to people. There is, however, another large area which is related to damage to property. From a standpoint of contamination, the difficulty may be associated with either the risk of injury to people who should subsequently occupy the premises or the risk of damage to materials such as film or instruments. It is even possible that a building may be decontaminated to the point where it is considered safe for human occupancy, but it still might be unsuitable for the processing of film or the manufacture of certain types of instruments.

The kinds of insurance with which we are concerned in the radiation field will fall into several categories. In the casualty business, workmen's compensation is related to reimbursing employees for injury resulting from their occupational exposures. Public liability insurance is related to the kind of injury or damage which could be done to the public as a result of industrial operations. Property damage insurance is related to damage or injury to the premises on which the operation may be located.

Perhaps the kind of insurance which is most susceptible to influence of radiation standards at the present time is workmen's compensation. The effect of a general lowering of radiation standards is fairly obvious. The lower the levels of permissible radiation, the smaller the chance that anyone would be injured. It should be pointed out, however, that there is no evidence that the kinds of levels which are being discussed have ever produced any direct injury to employees. This includes the 1934 values as well as those of today. All of the radiation damage in man of which we have any knowledge has been produced by levels of radiation many times those which have been set forth in the NCRP and ICRP recommendations. Thus, as the

values get lower and lower, chances of any injury become more and more improbable.

There are several factors, however, which may have a bearing on this problem. First of these is the possibility of delayed effects of exposures to radiation. These have bothered a great many people who are concerned about compensation problems because of the fact that the laws in some States require notification of accident or injury within a limited period following the injury. Second, the fact that some of the long-term effects such as leukemia, are also known to result from other causes. In most instances, the question of whether or not cases of such disease were radiation induced is virtually impossible to resolve.

Third, there has always been a tendency where standards have been available, even in the area of toxic materials, to utilize the standards as a basis of evidence of injury. This problem becomes more serious as one gets farther and farther away from the kinds of levels which are known to produce damage. As one approaches background levels of radiation, one encounters larger and larger numbers of people who receive these exposures. This would mean that we have more and more potential claims for compensation based solely on whether or not a standard has been exceeded.

Another, perhaps even more serious problem particularly among those who are engaged in radiation work, is the handling of individuals who have received exposures in excess of the maximum permissible doses or have accumulated a so-called lifetime dose. There is an increasing tendency on the part of industry to delve into the radiation history of persons who apply for employment. If an individual has accumulated a significant amount of radiation (excepting medical radiation) he may find himself in a position of being unable to obtain a job in the nuclear energy industry in which he may have had his entire industrial experience. Once again, the farther one gets below levels of known damage, the more acute this problem can become. These individuals will not be looking for compensation, they will be looking for a job. It is entirely possible that such people could be forced out of the nuclear energy field. I would like to stress again that the lower the NCRP values become, the more significant this problem will be.

The property damage field which includes, namely, contamination of the owner's premises and the problems associated with such contamination, will certainly provide the greatest potential source of financial loss. The contamination of an industrial installation can have two effects—one of these is the necessity of reducing radioactive contamination to levels which are safe for occupancy so that industrial operations can resume. It is also possible that under certain circumstances it may be possible to control the personnel exposure and still have the building unavailable for certain kinds of operations such as the processing of film, the manufacture of certain types of instruments or the processing of food or pharmaceutical products. These latter problems are not associated with NCRP recommendations, however, except where there may be a possibility of exposure to people. It does constitute a potential financial loss.

SUMMARY AND CONCLUSIONS

At the recent hearings of the Joint Committee on Atomic Energy in Washington, there was a considerable amount of discussion of the basis for establishing standards in the field of radiation hazard. While it was demonstrated, as we already know, that the values which are now being used are a result of detailed examination and evaluation of known animal and human data, a considerable amount of vital information is lacking. This is one of the reasons why it was repeatedly pointed out that maximum permissible doses should be used primarly as guides in radiation protection work. It was stressed frequently that these numbers should not be used as a specific borderline between what is safe and what is unsafe. The question was raised as to whether or not our standards are based upon a philosophy of risk. Inasmuch as we are rapidly approaching background levels in some of our standards, this question becomes more and more significant. If it is true, as has been asserted by some workers in the field, that any radiation is hazardous, then we can only assume that any radiation level above zero involves a risk of some type. Following this same line of reasoning, the farther one gets above zero, the greater the risk.

If a risk philosophy is accepted, it then becomes necessary to balance this risk against something in order to be able to determine whether or not the risk is worth taking. We are on the horns of a dilemma in attempting to solve this problem since we have not been able to establish the essential parameters. We do not know whether or not the curve of radiation damage vs. exposure goes through the zero point. We have not established how extensive the risk becomes as we deviate from zero. It would be helpful if we had some information relative to the consequences of over-exposure to radiation of various multiples of the maximum permissible dose. On the other side of the balance, we have to look at the benefits to be derived. This does not necessarily mean money. The benefits from the uses of radiation are many-fold and may include improvement of the health of our population, adding to our comfort and leisure by providing economic power or developing new products.

As the work of the NCRP goes on, it is to be hoped that with a refinement of our techniques in the accumulation of valid data on the biological effects of radiation on man, we may ultimately be able to solve the problem of whether or not we are, in fact, dealing continually with a risk situation. If we are, we should then devote more thought to an evaluation of the benefits to be bought for the price which we may be paying in the way of radiation injury. If, on the other hand, research and experience demonstrate that man, in fact, tolerates some level of radiation without injury of any kind, we will then and only then be able to establish meaningful, maximum permissible doses.

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V. THE FEDERAL RADIATION COUNCIL*†

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BACKGROUND

THE Federal Radiation Council represents something quite unique in the Federal Government organization. It provides for consideration at the highest level within the Government of a specific health problem, i.e. the effects of ionizing radiation on the citizens of this nation and for the development and promulgation of protection standards related to that problem. It is of interest therefore to examine the antecedents of the Council and the events leading to its formation.

The discovery and history of the uses of ionizing radiation have been very adequately described as has been the development of the various Federal programs in the radiation field. Members of the Health Physics Society have played an important role in these developments and I am sure are very familiar with them. Let us consider, therefore, recent events surrounding the development of the Federal Radiation Council.

On 3 April, 1959, the President announced that he had asked the Secretary of Health, Education, and Welfare; the Chairman of the Atomic Energy Commission; and the Director of the Bureau of the Budget to initiate a joint study of Federal organization of radiological health activities. The Bureau of the Budget, which in addition to its budgetary functions also serves as the President's staff for the improvement of management and organization in the Executive Branch, was directed to provide leadership for the study and to be responsible for making recommendations to the President when the study was completed.

Before considering the results of this study, let us examine the concept of the problem of radiation protection as it has evolved in recent years. Of primary importance in any health protection are the standards under which such programs operate, and here we must consider the invaluable contribution of the organization which in recent years has come to be known as the National Committee on Radiation Protection and Measurements.

In the early years of its operation, when it was known as the Advisory Committee on X-ray and Radium Protection, the NCRP examined the data on the effects of radiation exposure on man and attempted to define a level at which the obvious deleterious effects would not be seen. In other words, in the early days the philosophy was that one could determine a dose, often called a tolerance dose, below which damage would not occur. As more and more data accumulated there was increasing doubt that any such dose actually existed. Stated positively, it began to appear, at least in the case of the genetic effects of radiation, that there was no threshold dose; that any dose would be accompanied by some deleterious effects whether in the individual exposed or in future generations. Thus, the philosophy of the determination of radiation protection standards changed from one of determining a dose below which there would be no damage to determining a dose below which the damage would be acceptable, particularly in light of the obvious benefits which would be derived. To borrow a phrase from the Chairman

* Presented at the Fifth Annual Meeting of the Health Physics Society, Boston, Mass., 30 June 1960.

† This was presented at the Health Physics annual banquet rather than as part of the Symposium on the Effect of the NCRP Recommendations on National Life.

of the NCRP, Dr. LAURISTON TAYLOR, "the establishment of permissible levels of radiation exposure is not basically a scientific problem. Indeed, it is more a matter of philosophy, of morality, and of sheer wisdom . . . ."

The establishment of radiation protection standards presented itself as the central problem for which special considerations were necessary. Since the problem appeared so important and because the decisions on radiation protection standards involved passing judgment on how large a health hazard society is willing to accept for the benefits of radiation, it was felt that these decisions should be made at the highest level within the Government. Thus, on 14 August, 1959, the President established the Federal Radiation Council by Executive Order 10831 to, in the words of the official statement: "centralize responsibility for providing general standards and guidance to Executive agencies for their use in developing operating rules and regulations for radiological health protection". The Council was originally composed of the Secretary of Defense; Secretary of Commerce; Secretary of Health, Education, and Welfare; and the Chairman of the Atomic Energy Commission. The Secretary of Health, Education and Welfare was appointed Chairman of the Council, and the Special Assistant to the President for Science and Technology was authorized to attend the meetings and to participate in the deliberations of the Council.

During this time, an amendment to the Atomic Energy Act of 1954 was being developed to provide for the discontinuance of certain of the Atomic Energy Commission's regulatory responsibility and the assumption of these responsibilities by the States. A subsection was included in the Act to establish the Federal Radiation Council on a statutory basis. The law, in addition to adding the Secretary of Labor to the membership of the Council, provided for such other members as shall be appointed by the President. The Statute also provided that the Council "consult qualified scientists and experts in radiation matters, including the President of the National Academy of Sciences, the Chairman of the National Committee on Radiation Protection and Measurements, and qualified experts in the field of biology and medicine and the field of health physics . . . ".

The Statute described the function of the Council in the following language, "the Council shall advise the President with respect to radiation matters, directly or indirectly affecting health, including guidance for all Federal agencies in the formulation of radiation standards and in the establishment and execution of programs of cooperation with States. The Council shall also perform such other functions as the President may assign to it by Executive Order."

ACTIVITIES OF THE COUNCIL

The Council immediately set up a Working Group composed of senior technical representatives of the participating agencies. The Working Group has the responsibility of working on the development and conduct of the projects of the Council. Under the direction of the Council the Working Group deals with specific problems of Federal programs in radiation protection. The Group also anticipates problems which would require action by the Council. The Working Group members also serve as a co-ordinating point through which the competencies of the various agencies can be brought to bear on specific problems as they are considered.

Following its organization in the fall of 1959, the Working Group promptly directed primary attention to the problem of the development of recommendations by the Council on radiation protection standards. A working plan was developed directed towards the provision of initial recommendations on standards. The working plan called for extensive studies and consultations with scientists within and outside the Government. To expedite this study, the Council set up a tempory staff of four scientists made available by the participating agencies.

Thus, in the first few months of the existence of the Council several important principles of operation of the Council evolved. It was recognized that in order to provide recommendations on radiation protection standards the Council must consult with scientists both within and outside the Government. In these consultations it was necessary to consider the underlying assumptions and approximations upon which

recommendations of scientific groups on radiation protection standards were based, to review the most up-to-date information on the biological effects of ionizing radiation, and to elucidate some of the physical and chemical problems involved in the establishment and implementation of radiation protection standards. Is was also necessary to consult with staff members of the various Federal agencies concerned with radiation protection in order to define the problem areas to be provided for in recommended protection standards. In the preparation of the first staff report of the Council, consultations were held with some fifty scientists within and outside of the Government.

Another important principle reflected in the early work of the Council is the use of the *ad hoc* or temporary staff. The Federal Radiation Council does not wish to duplicate mechanisms, competencies, programs and so on, currently available within and outside of the Government. By assembling a small temporary staff for any given problem, the Council can obtain the services of people best qualified to deal with these special problems. By keeping this temporary staff small, the Council can pull together the data and information which is already available from a variety of sources and be provided with the best and most up-to-date information. Heavy reliance has been, and will continue to be, placed on the work of important groups such as the NCRP, the National Academy of Sciences, the American Standards Associations, and others in the field.

THE FIRST REPORT OF THE FEDERAL RADIATION COUNCIL

It has been repeatedly emphasized that the formulation of radiation protection standards involves a balancing of the risks to man of exposure to ionizing radiation against the benefits to be derived from the many important usages to which radiation is applied. The staff of the Council conducted a careful review of the current information on the hazards of ionizing radiation. As the recommendations were being developed, each agency examined the basic assumptions, the definitions, and recommendations in light of its statutory responsibility. Thus, the weighing of benefit versus risk was a continuous process during the evolution of the recommendations of the Council.

In this manner the Council prepared a Memorandum for the President on Radiation Protection Guidance for Federal Agencies. In this memorandum were seven recommendations. The recommendations were approved by the President on 13 May, 1960 and the memorandum was published in the *Federal Register* on 18 May 1960. The standards promulgated by the President are in the form of guidance to the Federal agencies. Operating standards and criteria will be developed by these agencies on the basis of this guidance. Also published at the same time, was the first staff report of the Federal Radiation Council entitled, *Background Material for the Development of Radiation Protection Standards*. This report is available from the Superintendent of Documents, U.S. Government Printing Office, Washington 25, D.C.

During the recent hearings before the Joint Committee on Atomic Energy there were extensive discussions on the background, the development and the content of these recommendations. It is certainly quite unnecessary in this paper to go over in detail all of the material concerning the Council's report that was covered in the recent Joint Committee hearings. For those who are interested, the transcripts of those hearings are available. The Joint Committee hearings have provided a valuable mechanism for a better understanding role of the Federal Radiation Council and its work to date.

There are several questions concerning the first report of the Council, however, which I think are of sufficient interest to justify mention here. In connection with the preparation of his testimony before the Joint Committee, the Chairman of the Council anticipated that certain questions might arise concerning the Memorandum for the President and the background document. Since these questions were of a scientific and technical nature, he asked the Working Group to prepare a memorandum answering the questions. Because I believe these questions and the answers are of particular interest, I would like to quote briefly from that memorandum.

"1. *How does one interpret the close agreement*

between the Federal Radiation Council recommendations and those of the National Committee on Radiation Protection and Measurements? It will be noted that the numerical recommendations developed by the Council do not differ substantively from certain other numerical recommendations such as those made by the National Committee on Radiation Protection and Measurements, the National Academy of Sciences, and the International Commission on Radiological Protection. This fact is highly significant. The Federal Radiation Council has made an extensive review of the many factors involved in the development of radiation protection standards. Many weeks have been spent in reviewing the basic biological data, for instance. The close agreement indicates that a careful review of all these data and factors involved in the formulation of radiation protection standards by different groups usually leads to the same conclusion. There are, of course, extensive areas of uncertainty, and the Council has indicated that it is urgent that additional research be undertaken and data developed to provide a firmer basis for evaluating biological risk. However, the degree of concurrence indicates that the data currently available point rather clearly to a consistent approach.

"2. *What is the meaning of Radiation Protection Guides?* The term Radiation Protection Guide has been adopted for Federal use. This term is defined as the radiation dose which should not be exceeded without careful consideration of the reasons for doing so; every effort should be made to encourage the maintenance of radiation doses as far below this guide as practicable. The term maximum permissible dose as used by the National Committee on Radiation Protection and Measurements and the International Commission on Radiological Protection is not being recommended for use by Federal agencies.

"There is more involved here than the arbitrary change of terminology. An important principle is involved. The term maximum permissible dose is often misunderstood. It has unfortunate and inaccurate connotations not intended by either the NCRP or the ICRP. The term Radiation Protection Guide with its definition more accurately represents the significance of numerical values in radiation protection programs. Because of the nature of radiation effects, and particularly the relationships between radiation dose and biological effect, there can be no single permissible or acceptable level of exposure without regard to the reason for permitting the exposure. Any numerical value is appropriate only for the circumstances for which it was developed. There can, and should be, different radiation protection guides with different numerical values depending upon the circumstances. The specific "Radiation Protection Guides" promulgated by the President by approval of the recommendations contained in the memorandum from the Federal Radiation Council are appropriate for normal peacetime operations.

"3. *How should the whole population* per capita *dose of whole body* 0.17 rem/year *be used?* For the individual in the population, the basic Guide for annual whole body dose is 0.5 rem. This Guide applies when the individual whole body doses are known. Under certain conditions, such as widespread radioactive contamination of the environment, the only data available may be related to average contamination or average exposure levels. Therefore, an operational technique is needed for assuring that the basic guide for individual exposure is being met. The Council suggests the use of the arbitrary assumption that the majority of individuals do not vary from the average by a factor greater than 3. Thus, where the individual whole body doses are not known, a suitable sample of the exposed population should be developed whose protection guide for annual whole body dose will be 0.17 rem *per capita* per year. It is noted that the use of the average figure as a substitute for evidence concerning the dose to individuals is permissible only when there is a probability of appreciable homogeneity concerning the distribution of the dose within the population included in the average. Particular care should be taken to assure that a disproportionate fraction of the average dose is not received by the most sensitive population elements. Specifically, it would be inappropriate to average the dose between children and adults, especially if it is believed that there are selective factors making the dose to children generally higher than that for adults."

FUTURE WORK OF THE COUNCIL

The first report was limited and considered only exposure guides. No specific recommendations were made concerning radioactivity concentration guides which is the term that has been substituted for the previously used maximum permissible concentration. Further reports will consider this problem.

The Federal Radiation Council recognizes the important work done by the NCRP, the NAS, the ASA and other groups. The Council has relied heavily on this work and will continue to do so in the future.

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LABORATORY WORK ON CERTAIN ASPECTS OF THE DEEP WELL DISPOSAL PROBLEM*

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Abstract—If there is sorption near the point of injection of the radioactive waste into a well, the concentration of activity per cubic foot may present a heat production problem due to radioactive decay. Cesium is a long-life fission product known to be easily sorbed by clays. The amount of cesium sorbed was found to be a function of several variables: (1) concentration of cesium in the waste; (2) concentration of other cations in the waste; (3) amount and nature of the clay present; (4) composition and amounts of other minerals in the formation; and (5) hydrogen ion concentration of the waste.

Illite sorbed 3.5 mg of cesium per 10 g of clay from a simulate 6 M neutralized Purex-type waste solution, while kaolinite sorbed 1.1 mg per 10 g and halloysite less than 0.5 mg of cesium. Increasing the sodium ion concentration from 0.5 M to 6.0 M decreased the amount of cesium sorbed from 1.02 to 0.39 mg per g of clay. Decreasing the cesium concentration from 100 mg/l. to 10 mg/l. resulted in decrease in the amount of cesium sorbed from 0.35 mg/g to 0.11 mg/g of clay. A mixture of small amounts of limestone with illite reduced the uptake of cesium from neutralized Purex waste. About three times as much cesium was removed from a sodium neutralized solution than from an acid solution.

The use of competing ions, like sodium and calcium, or the control of the Na/Cs ratio can limit the sorption of cesium within the range where localized concentration of heat will not represent a hazard to waste disposal through deep well injection.

INTRODUCTION

THE assumptions, facts, principles and problems of the deep well disposal program have been stated by DELAGUNA and BLOMEKE[1] and others. One such problem is fission product sorption near the point of injection of the radioactive waste into a well. The increasing concentration of activity per cubic foot of geologic material may present a heat production problem due to radioactive decay. BIRCH[2] of Harvard University has calculated the rates of heat production for two assumed formation thicknesses. The problem of safety which has to do with the generation of heat in the waste, was treated by BIRCH in terms of simplified models, which retain the principal features of the real situations, so long as the temperature remains within the same limits. These limits were then examined. The results suggest that nearly any quantity of waste producing heat at the rate of 0.01 W/gal, if distributed in a layer 100 m thick, could be accommodated without an undue rise of temperature. The rates were calculated for heat production and may be expressed as concentration of radioactive material per volume of formation. These rates correspond roughly to 2 and 8 c of strontium-90 or cesium-137 per ft^3 of formation.

It was the purpose of the laboratory experiments described below to determine the amount of sorption of radioactive cesium near the disposal well. The amount of cesium sorbed was found to be a function of several variables: (1) concentration of Cs in the waste; (2) concentration of other cations in the waste; (3)

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‡ Operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

amount and nature of the clay present; (4) composition and amounts of other minerals in the formation; and (5) hydrogen ion concentration of the waste.

INVESTIGATION

The sorption of cesium-137 by small amounts of clay was investigated. Several types of earth materials and clays, including illite, bentonite, montmorillonite, kaolinite, halloysite and Conasauga shale, were slurried with simulated 6 M Purex-type waste[3] which had been neutralized with sodium hydroxide. Purex is the nitric acid waste from a process by which the plutonium is recovered from irradiated natural uranium fuel elements. The procedure was to add from 0.3 to 20 g of clay to 300 ml of solution and stir for 2 hr at 216 rev/min, using a laboratory stirring device. After stirring, the solution was filtered through filter paper, and samples were taken for counting. The counting was performed on a well-type γ-scintillation counter using a NaI(Tl) crystal. Three samples were prepared for counting by placing 1-ml aliquots in 10 × 75 mm glass culture tubes. It was possible to determine the number of milligrams of cesium sorbed by the clay, knowing the initial and finial ratio of radiocesium to stable cesium. This same procedure was used in all the slurry tests.

Laboratory studies were made on the sorption of radiocesium on illite clay from solutions having varying sodium concentrations, Fig. 1. Concentration of Na ranged from 6 M to 0.5 M; this range of Na ion concentrations includes neutralized evaporated Purex waste, which is about 6 M and neutralized unconcentrated waste which is 1 M. The results show a threefold increase in cesium sorption with a tenfold decrease in sodium ion concentration.

Fig. 2 is a graph of various laboratory experiments showing the sorption of radiocesium from a 6 M sodium neutralized Purex solution by the clay minerals mentioned. The Conasauga shale contains a mixture of clay mineral and quartz. A sample of this shale analyzed by TAMURA[4] using X-ray, chemical and base-exchange procedures, was found to consist of

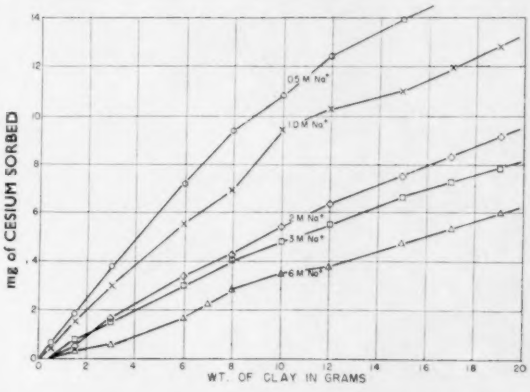

FIG. 1. The effect of sodium ion concentration on the sorption of cesium by illite clay. The concentration was 100 mg/l. in all tests.

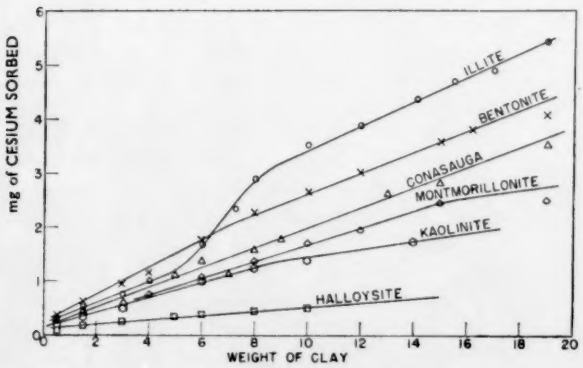

FIG. 2. Efficiency of various clay material for removal of Cs from neutralized Purex waste.

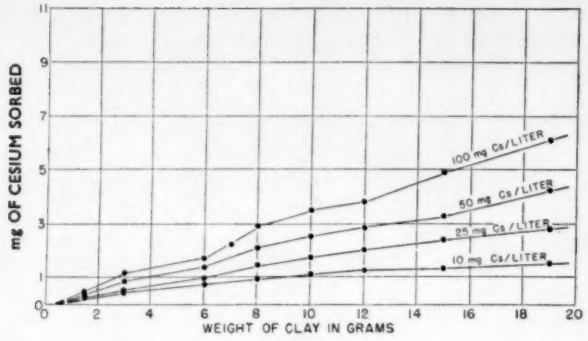

FIG. 3. Effect of cesium concentration on the sorption of cesium by illite clay from a 6 M $NaNO_3$ solution.

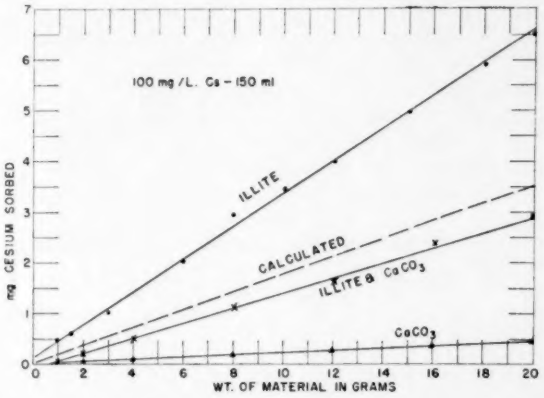

FIG. 4. The effect of a carbonate on cesium sorption by illite from a 6 M sodium neutralized Purex waste.

about 40 per cent illite, 25 per cent montmorillonite and 25 per cent quartz. Its ability to sorb cesium from the waste solution is large, considering the relatively large content of inert quartz. The bentonite sample is a clay of undetermined composition from Texas. As shown here, illite had the highest capacity and halloysite the lowest.

Also investigated was the effect of cesium concentration on cesium sorption from 6 M neutralized Purex-type waste, Fig. 3. The concentration studied ranged from 10 to 100 mg/l. of cesium. The 10 mg/l. cesium level is representative of undiluted 1 M Purex waste, while the 100 mg/l. level is slightly more cesium enriched than the 6 M evaporated waste. The results indicate approximately a threefold increase in cesium adsorption with a tenfold increase in cesium concentration. The relative effect on the amount of cesium sorbed per gram of clay, by increasing the concentration of sodium ion by the same factor, was the same.

Fig. 4 shows the amount of cesium sorption by illite and by 40–60-mesh limestone alone and in a 1:1 mixture with the clay. The small difference between the calculated values (dashed line) and the experimental results (solid line) shown on the two middle curves may be due to the small amount of limestone that does dissolve and thereby decreases the sorption capacity of the illite by that amount. These five test values tend to indicate that the presence of calcium carbonate or limestone which may be present in a disposal formation did not increase the sorption of cesium by illite clay.

In addition, a study was made on the amount of cesium sorption from 1 M nitric acid Purex waste. Fig. 5 shows the results on sorption of cesium from 1 M sodium and 1 M acid solutions containing 10 mg/l. of stable cesium carrier. These runs were made on both illite and Conasauga shale. The cesium sorption capacity of illite, in a 1 M acid waste solution is about half the capacity, compared to sorption from a

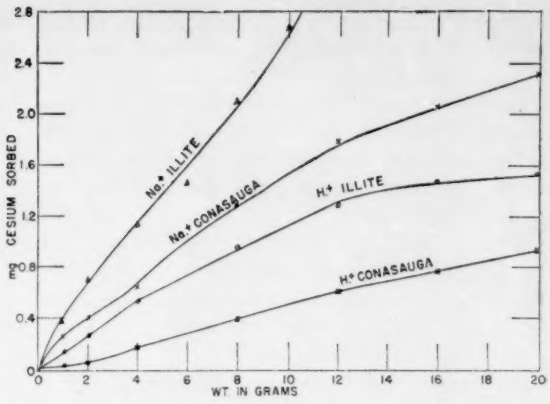

FIG. 5. Sorption of cesium from 1 M solutions of sodium and acid containing 10 mg/l. Cs carrier.

1 M sodium solution. Also, results indicated that the cesium capacity of Conasauga shale with 1 M sodium is decreased by almost 70 per cent when 1 M acid is used. There would be less cesium sorption, in a disposal formation on clay of this type, from an acid solution than from a neutralized solution.

To carry this one step further, it was known that sorption or ion exchange is a function of ionic radius. For exchange sites with no specificity, the tightness with which an ion is banded to a charge surface is an inverse function of the hydrated radius of the ion. However, cations must be dehydrated to enter the fixation sites. Cesium has an ionic radius of 1.65 Å and is more easily dehydrated than the lighter alkali metal cations. The presence of hydrogen ion depressed the sorption of cesium.

An attempt was made to indicate the relationship of the equilibrium studies reported to the expected results in a column operation. The method used involved a series of jar test equilibrium experiments where after 2 hr contact the solution was decanted and new solution was introduced to each beaker and reslurried. This same procedure of decantation was repeated and a third sample of the original solution added to the clay. The results from this experiment were compared to the saturation value determined, using a column of the same clay material and a large volume of the original solution. Using the exhausted clay material in the column as 100 per cent saturation it was determined that about 50 per cent of the saturation value was achieved by a single slurry treatment; the two successive slurry treatments resulted in an additional 60 per cent removal of the cesium present or about 80 per cent of the estimated concentrations of cesium remaining in solution.

Therefore, although more cesium could be sorbed from a fresh solution, one must increase the concentration of cesium to sodium or replenish the cesium already removed from the solution. The equilibrium value was not found to be changed by either increasing the contact time or the rate of slurry.

The results of these tests indicate that in a $NaNO_3$ system, the Cs/Na ratio of the solution determines the capacity of the clay for cesium removal. JACOBS and LACY[5] observed the same results in a series of studies on cesium sorption with Conasauga shale.

The most important mechanism of cesium removal from wastes by the clays and shale in these tests is that of ion exchange. The property of ion exchange results from a charge deficiency within the lattice of the clay mineral. Usually steric relationships preclude the penetration of the crystal lattice by counter-ions, resulting in the charge deficiency being effectively diffused over the surface of the particle. The charge deficiency is satisfied by the formation of an electric double layer at the interface between the crystal lattice and the solution.

The basal surface of the illitic fraction of the shale consists of open hexagonal networks of oxygen atoms. Successive lattices are stacked with their openings superimposed so that the void created can accommodate spheres with radii up to 1.64 Å without disruption of the

lattice. Cations of this approximate size, which move into these exchange positions, become tightly bonded or "fixed". In addition to the coulombic forces of attraction resulting from the charge deficiency in the crystal lattice, the cations in these positions are held by the formation of covalent bonds with the surrounding twelve oxygen atoms.

As a result of the cesium sorption by ion exchange and coulombic forces and the dependence of the mechanism on other parameters, the following conclusions were made: (1) sorption of cesium is a function of the Na/Cs ratio; (2) limestone very slightly, if at all, decreases the amount of cesium sorbed per gram of illite; (3) amount of cesium sorbed under jar test or equilibrium conditions, which are reached very rapidly, is about 50–60 per cent of the saturation value obtained using a dynamic system, i.e. columns or cores; (4) increased H ion concentration decreases sorption.

These results suggest that based on the reasonable heat capacity values stated earlier, undue localized heat production, due to sorption of radioactive cesium-137, near the well or within the formation, can be controlled so that it will not be a serious problem. The disposal of diluted radioactive waste in the porous strata of deep (a few thousand feet) synclinal structure would seem safe based on heat production due to sorption of radiocesium.

Acknowledgments—I would like to thank Mr. O. M. SEALAND and Drs. TAMURA and JACOBS who materially assisted in this work, and also the members of the Waste Disposal Research & Engineering Group, H.P. Division, Oak Ridge National Lab., Oak Ridge, Tennessee, for their many helpful suggestions.

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A SENSITIVE POCKET DOSIMETER READER*

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Abstract—A modified Townsend balance circuit was used to measure the discharge of pocket ionization chambers. The reader consists of a vibrating reed electrometer, a recorder and a stable voltage supply. This system has good sensitivity and excellent linearity. Also, because of negative feedback, the reader is nearly independent of the capacity of ion chambers that have the same volumes and that are made of the same materials. The discharge of a chamber with a volume of 5 cm^3 exposed to 1 mr of radiation can be reproduced to ± 1.5 per cent.

INTRODUCTION

SEVERAL pocket pencil dosimeter readers have been developed at Hanford. A pencil reader with good sensitivity and fair linearity was invented by ROESCH *et al.*[1] Another reader has been developed which employs a TOWNSEND balance circuit.[2] This circuit in its simplest form is described by HINE and BROWNELL.[3] The essential feature of the circuit is that the final result is not influenced by the input capacity of the system.

One of the first automatic Townsend balances was developed by SCHERBATSKOY and FEARON[4] in 1946 (Fig. 1). This automatic system used a new type of vibrating reed electrometer that employed a negative feedback loop in the circuitry. In this instrument the vibrating capacitor produces an a.c. voltage which is amplified, rectified and then fed back. An analysis of this circuit results in

$$\Delta Q = E_f \left[\frac{C_d}{G} + C_c \left(\frac{1+G}{G} \right) \right] \qquad (1)$$

where ΔQ = charge to be measured;
C_d = average dynamic condenser capacity;
C_c = capacity of the feedback condenser;
E_f = feedback voltage;
G = gain of the amplifier.

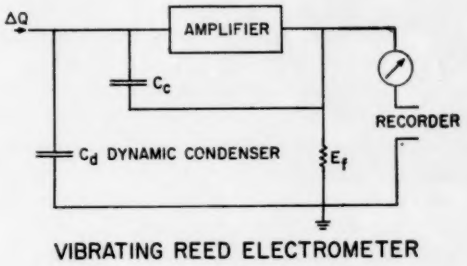

FIG. 1. Automatic Townsend balance circuit.

Since the gain of the system is nearly 1000,

$$\Delta Q \approx E_f C_c \qquad (2)$$

Here is another example of the Townsend balance circuit, since the charge is determined by measuring C_c and E_f and is nearly independent of the input capacity C_d and any capacity that is in parallel to C_d. This instrument was improved by PALEVSKY *et al.*[5]

KIRN and KENNEDY[6] were the first to use the vibrating reed electrometer to measure the discharge of pencil dosimeters. The circuit described in this paper is similar to the one used by KIRN and KENNEDY. Certain features have been added to make the instrument more convenient and more sensitive.

PENCIL READER

The pencil reader (Fig. 2) consists of a vibrating reed electrometer, a stable power supply

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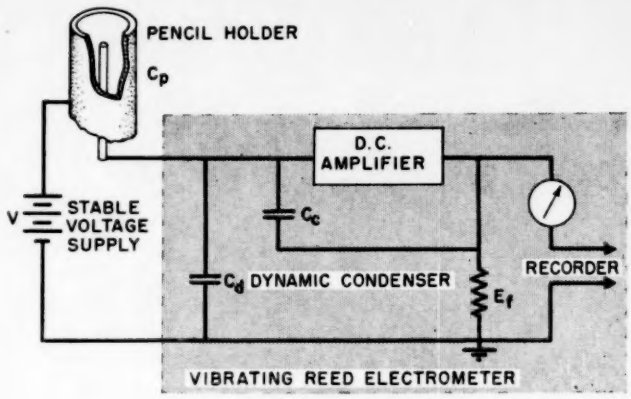

FIG. 2. The pencil reader.

and a recorder. Notice that these instruments are available in many physical laboratories. In operation the pencil is placed in the holder and charged by voltage supply V. The chamber is then taken to the field and discharged by an amount ΔV. In recharging, the electrons are drawn from condensers C_d and C_c. The dynamic condenser produces an a.c. voltage which is proportional to ΔV. This voltage is then amplified, rectified and fed back. The feedback voltage is read either on the electrometer meter or more conveniently on a recorder. Because of the high input impedance of the electrometer, the charge does not leak off and the feedback voltage remains as a constant reading on the recorder chart.

An analysis of this circuit yields

$$\Delta Q = E_f\left[\frac{C_p}{G} + \frac{C_d}{G} + C_c\left(\frac{1+G}{G}\right)\right] \qquad (3)$$

where C_p is the capacity of the pencil and the pencil holder. Notice that equation (3) is the same as equation (1) except for the term containing C_p. Since C_p is also divided by G,

$$\Delta Q \simeq E_f C_c \qquad (4)$$

This means that pencils of the same cavity size and material do not have to be calibrated separately. Another feature of this reader is that the pencils are charged and read in one operation.

EXPERIMENTAL

Victoreen model 362 pencils were used to check the performance of the pencil reader. Out of forty pencils, two were found that would reproduce 1 mr to ± 1.5 per cent. Five other pencils out of this same group of forty could reproduce an accumulated dose of 1 mr to ± 3 per cent. The rest of the pencils would reproduce an accumulated dose of 1 mr to ± 5 per cent after suitable tightening of the insulators. The above uncertainties are standard deviations for a single reading. In a group of ten readings the more accurate pencils give a standard deviation from the mean of ± 0.50 per cent. In tightening, static charges and strains are set up on the insulators. These charges and strains can be partially removed by exposing the pencils to several hundred roentgens of γ-radiation.

A linearity curve is shown in Fig. 3. The average deviation of the points from a straight line is $\frac{3}{4}$ per cent.

DISCUSSION

The vibrating reed electrometer used in this work was an Applied Physics Corporation model

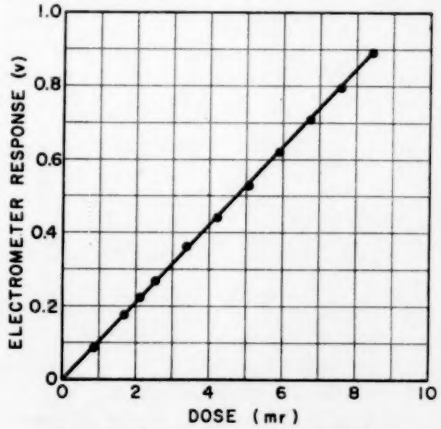

FIG. 3. Calibration curve for Victoreen model 362 pocket ion chambers.

30 with a full-scale voltage of 1 V. This permits measurement to 10 mr with the model 362 pencil. This range can be extended by either replacing condenser C_e with a larger condenser or using an electrometer that has a larger voltage range. With this improved reader one should be able to expose a 1 cm^3 chamber to 1 mr and obtain a standard deviation from the mean of ten readings of 2.5 per cent. This size chamber should be useful for depth dose measurements and other biological experiments both in γ-ray and neutron dosimetry.

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SILVER METAPHOSPHATE GLASS FOR X-RAY MEASUREMENTS IN COEXISTENT NEUTRON AND γ-RADIATION FIELDS

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Abstract—Small glass rods, 1 mm in diameter by 6 mm long, composed of the metaphosphates of Al, Ba, K and Ag, were exposed to both neutrons and γ-radiation to determine their suitability as γ-dosimeters in γ-radiation fields coexistent with fast neutrons. Filters in the form of cylindrical capsules were constructed to eliminate the major portion of the increased response per unit of exposure dose in the photoelectric region. The fast neutron response was determined in the energy range of 0.5 to 1.5 MeV; the maximum response in this range of energies was 0.7 per cent of that of Co^{60} γ-radiation in units of tissue rad. Thermal neutron exposures were made in the ORNL graphite reactor thermal column; flux measurements were made with Au and Au + Cd foils. A response equivalent to an exposure dose of 1 r of Co^{60} γ-rays was induced by $(2.96 \pm 0.38) \times 10^9$ n_t/cm^2. Preliminary measurements with similar rods in which the elements Ba and K had been replaced by Mg and Li, respectively, indicate a lower peak γ-ray response per unit of exposure dose in the photoelectric region and an increased sensitivity to thermal neutrons; 1 r equivalent response was produced by $(3.7 \pm 0.4) \times 10^8$ n_t/cm^2.

I. INTRODUCTION

RECENT studies on the "radiophotoluminescence" properties of silver-activated metaphosphate glass[1–4] indicate new possibilities in the development of radiation dosimeters; this paper is the description of a method using this glass to measure γ-ray dose in γ-radiation fields coexistent with neutrons. "Radiophotoluminescence" has been studied by a large number of investigators[5] and has been defined as a phenomenon whereby a material, originally non-luminescent under ultraviolet or visible light, is made responsive to such excitation by pretreatment with γ-rays or X-rays.

Preliminary studies and the reported characteristics of silver metaphosphate glass[1–4] showed that it might be an important tool in γ-ray dosimetry in fields of γ-rays and neutrons. The objectives of this investigation were: (1) to determine the γ-response per unit exposure dose as a function of photon energy; (2) to determine the response per first collision tissue rad of fast neutrons as a function of neutron energy; (3) to determine the response per unit flux of thermal neutrons; and (4) to investigate methods of eliminating or correcting for response characteristics which would adversely affect the measurement of γ-radiation exposure doses in mixed fields of neutrons and γ-rays.

II. RADIOPHOTOLUMINESCENCE AND SILVER METAPHOSPHATE GLASS

In alkali halides the electrons are ejected from the negative halide ions as a result of irradiation, and then travel through the crystal until captured at vacant anion positions.[6–10] These trapped electrons, called "F-centers", are capable of absorbing visible light, the absorption coefficient of which is a measure of the radiation dose received. However, in most alkali halide crystals, the thermal bleaching rate of the induced F-center coloration is too rapid to permit their use as dosimeters.

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† Operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

The fluorescent properties of several materials are markedly enhanced by the introduction of some impurity substance whose atoms occupy interstitial positions and act as traps for electrons liberated by radiation.[6] One of the most promising materials for satisfying the demands of present-day γ-dosimetry is glass activated with silver as an impurity.[9] When such glass is exposed to radiation, loosely bound electrons are released from the negative ions of the glass and some of them are trapped by interstitial silver ions, forming a special type of F-center, or "luminescent center". In the capture of liberated electrons, the Ag^+ ions are in competition with electron "traps" which arise from irregularities (i.e. impurities) in the molecular structure of the glass. The traps provide alternate energy levels which can be occupied by the electrons. The distribution of electrons between Ag^+ ions and traps will, of course, control the fluorescence–dose relations of the particular specimen of glass. When these newly formed neutral Ag^0 atoms (metastable) are exposed to ultraviolet light, they become excited through the absorption of energy and, upon returning to the ground state, they release fluorescent radiation of about 6400 Å wavelength.

SCHULMAN *et al.*[1] found that metaphosphate glass containing $AgPO_3$ as an impurity had the greatest fluorescent efficiency when the mixture contained 8 per cent $AgPO_3$. The base glass components are $Al(PO_3)_3$, 50 per cent, $Ba(PO_3)_2$, 25 per cent, and KPO_3, 25 per cent. Between 0°C and room temperature, SCHULMAN *et al.*, found that there was a 24-hr time lapse after exposure before the fluorescent response reached its maximum value.

III. FLUORESCENCE EVALUATION TECHNIQUE

The mechanism by which fluorescence is produced in silver-activated glass was described in Section I above. By exciting the glass with 3650 Å ultraviolet light the fluorescence can be maintained while dose determinations are made. The intensity of the fluorescence depends upon the radiation dose and the number of silver atoms receiving excitation energy from the ultraviolet light. Hence, for a given radiation dose, the measurable fluorescence depends upon the amount of excitation energy absorbed and the amount of fluorescent light reaching the phototube. Small glass rods, 1 mm diameter by 6 mm in length, of the type reported by SCHULMAN and ETZEL[11] and by ETZEL *et al.*[12] were used in this study; they were read in a commercially available fluorimeter.[13] Another type of fluorimeter is described in Refs. 11 and 12. Slightly different versions of fluorimeters have been used by DEGELMAN *et al.*[14] and by KONDO[15]. In the fluorimeter reader, ultraviolet light (3650 Å) passes through a simple optical system to the end of the glass rod. At one point the light impinges on a mirror where a small part of it passes through a slit to a reference phototube. The fluorescent light resulting from the absorbed ultraviolet radiation is then gathered by a conical reflector (a part of which is the chuck holding the rod) and directed by this reflector through an orange-pass filter to the fluorescence detecting photomultiplier.

The two phototubes are arranged in a null system and permit a measurement of the ratio of the intensity of the exciting radiation to that of the fluorescent radiation. A null reading indicates the balance of two triodes in a comparitor with the microammeter between their plates reading zero. In operation the circuit is first balanced without the glass rod in the light path. When the glass rod is moved into the ultraviolet light path a positive reading of the meter needle indicates the presence of any fluorescence from the excited silver atoms in the rod. The rods are read before and after being exposed to radiation, and the difference of the pre-dose and post-dose readings is proportional to the dose received.

IV. CALCULATION OF X- AND γ-RAY RESPONSE

It is assumed that the glass response is proportional to the energy deposited in it by electrons. To calculate the energy absorbed in the glass, assume a monoenergetic beam of photons of initial intensity I_0 in MeV/cm^2, incident on the glass and a subsequent arbitrary transmitted monoenergetic intensity I. Under narrow beam geometrical conditions, assuming

that the presence of the small glass rod does not perturb the field significantly, the energy absorbed per roentgen exposure is inversely proportional to $[(\tau + \sigma_\alpha)/\rho]_{\text{air}}$, the true mass absorption coefficient for air in cm^2/g for energies up to 3 MeV (τ = photoelectric cross-section, σ_α is Compton absorption cross-section, and ρ is the density in g/cm^3).

The energy absorbed in the glass, E_a, was calculated for photon energies ranging from a few keV to 1 MeV. The response R per roentgen exposure for photons of energy E_γ relative to the response for 1.0 MeV photons is given by:

$$R = \left\{\frac{E_a}{[(\tau + \sigma_\alpha)/\rho]_{\text{air}}}\right\}_{E_\gamma} \div \left\{\frac{E_a}{[(\tau + \sigma_\alpha)/\rho]_{\text{air}}}\right\}_{E_\gamma = 1.0\ \text{MeV}}$$

The result of this calculation is shown in Fig. 1. The experimentally determined response relative to Co^{60} response is also shown. The response to Co^{60} γ-radiation was measured under electronic equilibrium conditions as were the responses to X-rays. Since the response calculated in the above manner assumed that the secondary electrons produced by 1.0 MeV photons do not escape from the glass, only an adjustment in the calculated response to correct for the X-ray spectrum must be made. This was done using VILLFORTH *et al.*'s calculation[16] of KRAMER's distribution for filtered X-rays at the energies used and subsequent normalization of the results to unit dose. The middle curve in Fig. 1 shows the response thus corrected (all values in Fig. 1 were normalized to the 200 keV response).

An attempt was made to correct the adverse energy dependence of the glass in the photoelectric region. A unit response per unit exposure to an accuracy of ± 30 per cent was achieved for energies greater than 115 keV using a 0.020 in. tantalum cylinder and 0.030 in. tin around 0.015 in. Teflon. This capsule was enclosed on each end with 0.010 in. tantalum. The photon energy dependence with such an arrangement is negligible over the fission γ-spectrum. The glass response as a function of photon energy is shown in Fig. 2 for exposures in several capsules. The discrepancy at the K-edge of tantalum is believed to be caused by the width of the spectrum of the X-rays used in the experimental exposure.

V. RESPONSE OF THE GLASS TO FAST NEUTRONS AS A FUNCTION OF NEUTRON ENERGY

The total neutron cross-section in the energy range from a few keV to several MeV is approximately equal to the elastic cross-section for the

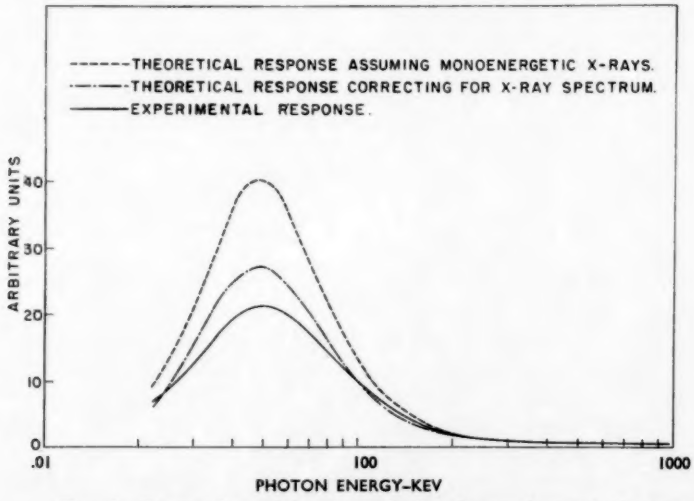

FIG. 1. Comparison of the theoretical and experimental photon response of the unshielded glass.

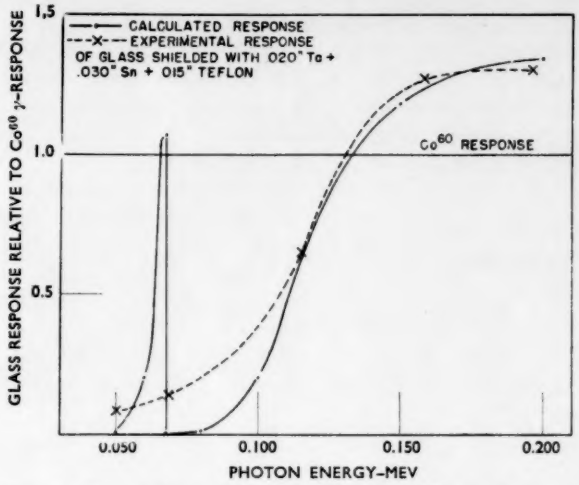

FIG. 2. Comparison of theoretical and experimental γ-responses of the shielded glass.

elements in the glass. Therefore, essentially all of the energy deposited in the glass by fast neutrons, in this energy range, is imparted by elastic collisions. On this basis, the energy absorbed per gram per n/cm^2, assuming the dimensions of the rod to be large relative to the path length of the recoil atoms, is 0.125×10^{-7} erg g^{-1} per n cm^{-2} for 1.5 MeV neutrons. Determination of the energy absorbed in tissue for 1.5 MeV neutrons gives 2.7×10^{-7} erg g^{-1} per n cm^{-2}. Comparison of these values shows the glass absorbs 4.6 per cent as much energy as tissue from a given integrated flux of 1.5 MeV neutrons.

The fluorescent response of glass per glass rad of fast neutrons is less than that per glass rad of γ-radiation; this indicates a lower effectiveness of heavy ions, relative to electrons, for producing fluorescent centers. The average effectiveness of neutrons in the 0.5 to 1.5 MeV range is about 0.12 relative to γ-radiation.[17] KONDO has suggested an RGE (relative glass effectiveness) factor for this.[18] His measurements for 14 MeV neutrons indicate an RGE of less than 0.25. Although the value of RGE for 14 MeV neutrons is approximately twice that for 1.5 MeV, the quantitative relationship of RGE and neutron energy has not been established.

The experimentally determined equivalent γ-response of the glass as a function of neutron energy is shown in Fig. 3. These measurements

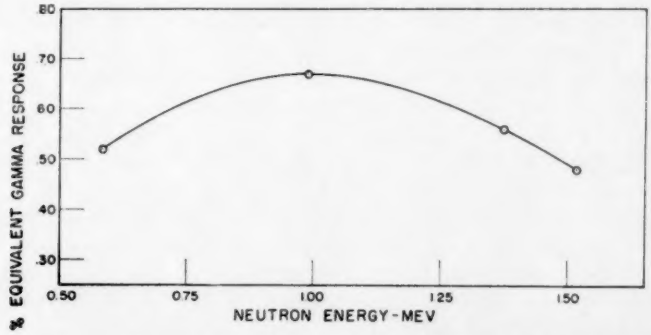

FIG. 3. Neutron response of glass relative to Co^{60} γ-response.

were made using the ORNL 3 MeV Van de Graaff to produce neutrons in a tritium gas target by the $H^3(p,n)He^3$ reaction. The response per tissue rad relative to the response per tissue rad of Co^{60} γ-rays varies from 0.5 per cent at 1.5 MeV to 0.7 per cent at 1 MeV.

VI. RESPONSE OF THE GLASS TO THERMAL NEUTRONS

For the metaphosphate glass the principal thermal neutron absorbers are Ag^{107} and Ag^{109}, with cross-sections of 30 barns and 84 barns, respectively. The nuclei resulting from neutron capture, Ag^{108} and Ag^{110}, are β- and γ-emitters with short half-lives of 2.4 min and 24 sec, respectively; this indicates that there will be no prolonged buildup of dose after irradiation has ceased. Owing to the small size of the glass rods, most of the γ-rays escape without being absorbed, and hence it may be assumed that the major portion of the response to thermal neutrons is induced by the β-radiation. The response resulting from irradiation by thermal neutrons should be directly proportional to the neutron flux to which the glass was exposed. The measurement of γ-dose in a mixed field might be accomplished by encasing the glass rods in a neutron absorber which would not emit β-rays or γ-rays. Li^6, with an (n,α) cross-section of 945 barns, serves well as a thermal neutron shield material.

Three sizes of encapsulating cans, each containing a different thickness of Li^6 metal, were constructed. The relative sizes of the cans and the dimensions of the lucite box in which exposures in the thermal column were made are indicated in Fig. 4. All three cans had a cavity of the same size and aluminum walls of the same thickness, but the thickness of Li^6 metal varied from $\frac{1}{4}$ in. for can A to $\frac{1}{8}$ in. for Can C.

The source of thermal neutrons was the thermal column of the ORNL graphite reactor.

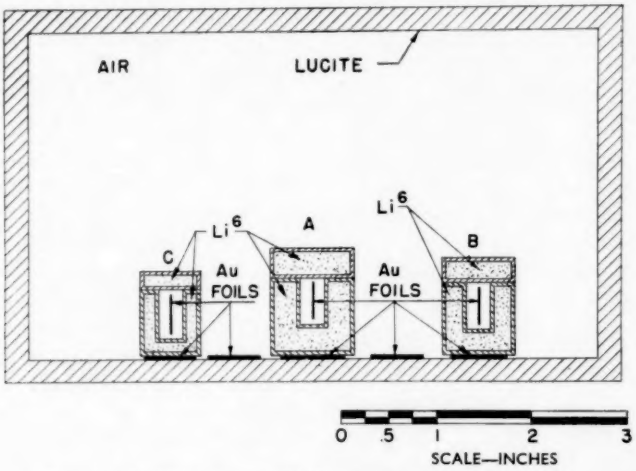

FIG. 4. Lucite box containing Li^6 cans and Aufolis.

The flux in the region where the exposures were made was approximately 5.25×10^6 n_{th}/cm^2 sec for a reactor power level of 3400 kW. Since Li has a high absorption cross-section for thermal neutrons it was necessary to measure the flux density distribution in the vicinity of the cans and inside the cans to determine the attenuation factor of each can. These flux density relationships were determined by placing gold and Au + Cd foils inside and outside the cans, when the latter were positioned in the lucite box. The sealed box was lowered to the center of the bottom of the thermal column for exposure. The flux density values thus obtained are listed in Table 1. From these data it was shown that the Li can C reduced the thermal

flux by a factor of 100, can B by a factor of 300, and can A reduced it by a factor of 1100. For general use, can C, which has a thickness of $\frac{1}{8}$ in. of Li^6, is adequate.

For the exposure of the unshielded glass the dosimeters were encased in fluorothene capsules (2.5 mm wall thickness) to insure electron equilibrium. They were then suspended in the air-filled lucite box at the positions occupied by the sample cavities of the Li cans as indicated in Fig. 4. Exposures were made at the center of the bottom of the thermal column.

To provide γ-energy independence for exposures inside the Li^6 cans, the glass rods were encased in the 0.020-in. tantalum capsules described previously. These capsules could not be used in the neutron field because tantalum becomes activated by thermal neutrons and thus results in a higher response in the glass. Using a 50-cm^3 graphite shell ion chamber, BAGLEY *et al.*[19] measured the γ-dose rate at the position of these exposures and obtained a value of 8.0 r/hr at a power level of 3400 kW. The response of the glass, with and without the Li^6 shield, indicated an equivalent exposure dose rate of 7.94 and 14.32 r/hr, respectively. The difference, 6.38 r/hr, is the roentgen equivalent response induced by the integrated thermal neutron flux. An integrated flux of $(2.96 \pm 0.38) \times 10^9$ n_t/cm^2 produced a response equivalent to one roentgen exposure of 1 MeV γ-radiation. This value corresponds closely with the value of 3.0×10^9 n_t/cm^2 calculated by KONDO[18] from exposure data obtained at the thermal column of the Brookhaven reactor. However, it is considerably lower than the value of 1.1×10^{10} n_t/cm^2 suggested by the Harwell Research Group.[4]

Table 1. Thermal neutron flux values inside and outside Li^6 cans

Foil position	Flux (n_t/cm^2 sec)
Underneath can A	2.44×10^6
Underneath can B	2.74×10^6
Underneath can C	2.85×10^6
Inside can A	2.16×10^3
Inside can B	8.85×10^3
Inside can C	2.05×10^4
Between can A and B	2.21×10^6
Between can B and C	4.39×10^6

A Li^6 shield of $\frac{1}{8}$ in. thickness is sufficient to absorb 99 per cent of the thermal neutrons and make it possible for a silver-activated metaphosphate glass rod to serve as a γ-ray dosimeter in a mixed field of neutrons and γ-rays. If the integrated thermal neutron flux is known, the quotient obtained by dividing this flux by 2.96×10^9 n_t/cm^2 per equivalent roentgen is the thermal neutron induced equivalent response.

VII. LOW-Z GLASS PROPERTIES

Bausch and Lomb Optical Company has recently made available a silver-activated metaphosphate glass of the type described by GINTHER and SCHULMAN[20] in which the elements Mg and Li have been substituted for Ba and K, respectively. The glass contains these lower atomic number elements in the naturally occurring isotopic abundance ratio, i.e. the Li contains about 7.4 per cent of Li^6. Since Li^6, when exposed to thermal neutrons, undergoes a (n,α) reaction, the number of electrons available to form "luminescence centers" is increased due to the ionization by the α-particles. It would be expected that this "low-Z" glass, when exposed to the mixed radiation field in a reactor thermal column, would exhibit a higher response, hence it might serve as a sensitive detector for thermal neutrons. However, the primary purpose of substituting "low-Z" elements in the glass is to lower the photoelectric absorption in the X-ray range (22 keV to 200 keV) and thus reduce the energy dependence peak in the 50 keV region.

"Low-Z" glass was not available in sufficient quantity to permit a large number of exposures. Consequently, the results presented here are based on preliminary measurements only. The results of a preliminary study of the dependence of the response on photon energy in the X-ray range are shown in Fig. 5 with the response normalized to the Co^{60} γ-response. The peak of the curve, compared to that for the regular or "high-Z" glass (Fig. 1), is lower by approximately a factor of 2.

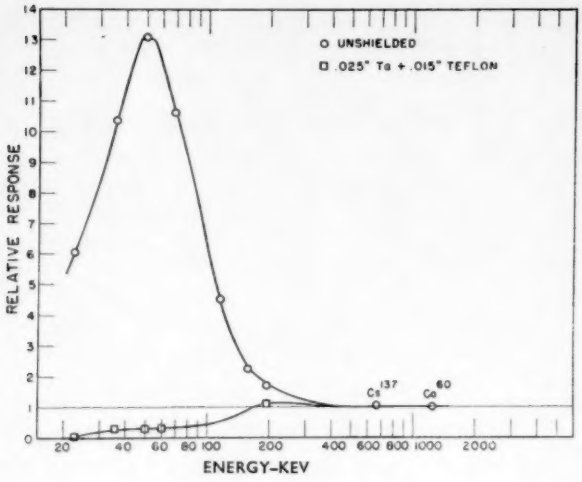

FIG. 5. Low-Z glass response as a function of photon energy.

Exposures in the thermal column similar to those for "high-Z" glass and computations, as above, yielded a value of $(3.6 \pm 0.05) \times 10^8$ n_t/cm^2 to produce a fluorescence response equivalent to that produced by 1 r of 1 MeV γ-radiation.

VIII. SUMMARY

Silver metaphosphate glass exhibits desirable characteristics as a γ-dosimeter for a wide range of applications, including nuclear accident monitoring. After exposure to X- or γ-radiation, the glass fluoresces at about 6400 Å when excited by 3650 Å ultraviolet radiation. The fluorescence response is independent of exposure dose rate and exhibits a linear relation to total exposure dose; for the 1 mm $\times$ 6 mm rods and the reading technique used, the linear relationship for exposure dose vs. response holds to about 10,000 r. Encapsulating filters, suitable for many applications, have been made to correct for the increased response per unit exposure dose in the photoelectric region.

For γ-ray fields coexistent with fast neutrons the glass may be used, with the Bausch and Lomb fluorimeter, for integrated exposure doses above 5 r. In the portion of the fission neutron spectrum including the greatest number of neutrons, the response to one tissue rad of fast neutrons is less than 1 per cent of that for one tissue rad of γ-rays.

A thermal neutron integrated flux of $(2.96 \pm 0.38) \times 10^9$ n_t/cm^2 produces a fluorescence response equivalent to an exposure dose of one roentgen of γ-radiation. Small encapsulating shields containing Li^6 were constructed to permit measurements of γ-rays coexistent with thermal neutrons.

The glass may be read many times without detectably affecting the absolute value of the response. Therefore, it may be used as an "extended interval" personnel dosimeter for γ-rays. The response of such a dosimeter is a sensitive function of photon energy unless properly encapsulated. A separate measurement of thermal neutron flux, such as with Au and Au + Cd foils, permits an approximate correction for the thermal neutron response in mixed radiation field dosimetry when Li^6 shields are not used.

Preliminary studies with special "low-Z" glass indicate that it has a greater response to thermal neutrons and a lesser relative response to X- or γ-radiation in the photoelectric energy region than the regular glass.

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INVESTIGATIONS ON THE RELATIVE β- TO γ-RAY SENSITIVITY OF PHOTOGRAPHIC EMULSIONS

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Abstract—The dosage sensitivity of two conventional dosimeter films was found to be 10 per cent greater for Sr^{90}–Y^{90} and U β-rays than for Co^{60} γ-rays when an equivalent amount of energy was expended in the emulsion. These differences did not appear when a nuclear emulsion was substituted for the dosimeter films. Results are explained on the basis of the inherent grain sensitivities of the different emulsions.

INTRODUCTION

THE response of photographic film to β- rays and electrons is well established. FLEEMAN and FRANTZ[1] have shown that film blackening for electrons is directly proportional to dose from 0.5 to 1.4 MeV. The study was extended by DUDLEY[2] to include low-energy β-rays and electrons. Film sensitivity was found to decrease markedly with lower energies, which was primarily governed by the range of the particles and their ability to penetrate the entire thickness of emulsion. In studying the β-ray sensitivity of personnel-monitoring film TOCHILIN and GOLDEN[3] concluded that film response is constant for β-ray energies greater than 2 MeV and that, within these limits, essentially the same density will be produced by either 1 rad of β- or γ-rays. Similar conclusions have been reached by other investigators.

Subsequent studies by TOCHILIN *et al.*[4] relating film response to specific ionization indicated that film should be more sensitive to high-energy β-particles than to the secondary electron distribution from γ-rays normally used for calibration of film dosimeters. The present report investigates this hypothesis for several different films, including one nuclear emulsion.

EXPERIMENTAL TECHNIQUES

β-Rays used for this study were from a uranium disk, a source widely used as a calibration standard, and from Sr^{90}. Two Sr^{90} sources were used, the first being a point source, the second being a disk made by mixing a fixed amount of the isotope with dental plaster. Absorbers placed over both of the Sr^{90} sources selectively attenuated the 0.6-MeV Sr^{90} component and limited exposure primarily to the 2.2-MeV β-rays from the Y^{90} daughter.

β-Ray spectra were determined with an anthracene crystal and a 100-channel scintillation spectrometer. It was found that the spectrum from the uranium disk could be matched to the Sr^{90} disk by filtering with 80 mg/cm^2 of plastic absorber.

β-Ray pulse-height spectra obtained for the filtered and unfiltered Sr^{90} disk are shown in Fig. 1 together with the matched uranium spectrum. The curves for Sr^{90} illustrate the effective removal of the low-energy component with 107 mg/cm^2 of plastic absorber. Some degradation of the 2.2-MeV Y^{90} component is also apparent. From the measurements of BROWNELL[5] on P^{32} β-rays it appears that the degradation at high energies could be minimized by using an absorber of high atomic number.

Table 1 lists the measured exposure rates for the three sources obtained with an extrapolation chamber. The chamber sensitivity did not allow a direct measurement of β-radiation from the uranium disk. However, the total rate of α- plus β-ionization at the polished surface was 14.4 e.s.u./cm^3 per hr, in complete agreement with earlier measurements of BORTNER[6]. BORTNER's value of 0.240 e.s.u./cm^3 per hr was then taken as the β-ray dose rate with a plastic absorber of 7 mg/cm^2 covering the source. The dose rate for an 80 mg/cm^2 absorber was obtained by taking relative readings with a surface β-ray dosimeter for the two absorbers. The conversion to rads per hour was derived from calculations by MORGAN[7] that 1 e.s.u./cm^3 of β-ray ionization measured in air is equivalent to 95 ergs of energy absorbed per gram of average wet body tissue.

In measuring γ-ray exposure with ionization

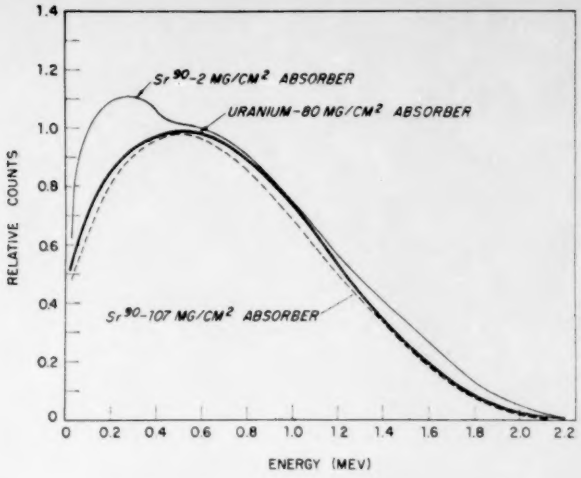

FIG. 1. β-Ray pulse-height spectra from uranium and Sr^{90}.

chambers it is established that ionization increases with wall thickness. Equilibrium is attained when wall thickness becomes equal to the maximum range of the secondary electrons produced in the medium. The necessary conditions for measuring roentgen dose are then achieved. GREENING[8] has shown that films should also be surrounded by an equilibrium layer to establish a maximum electron flux through the emulsion. This analogy is complicated by the additional requirement that the film be shielded with a material of high density such as Ag, Cd or Pb in order to minimize energy dependence from low-energy X-rays.

The influence of Ag and Pb filters on film blackening was investigated with Co^{60} γ-rays, first by exposing the emulsion in direct contact with the filters and then by interposing increasing thicknesses of paper absorbers between the two. Ag and Pb foils 0 040 in. and 0.030 in. thick were used; however, foils ranging from 0.010 to 0.060 in. produced negligible change in sensitivity. Curves for the effect of forward- and back-scattered electrons on film are shown in Fig. 2. Electron emission from the surface of the foils is seen to be strongly dependent on the atomic number of the filter material and on the direction of the secondaries. These results are explained in a detailed study made by HINE[9] on the absorption of γ-rays in various materials and the associated production of secondary electrons. The points of greatest importance are that film sensitivity is influenced both by the filter and the manner in which the film is packaged. This effect is minimized if an equilibrium thickness of paper or other "tissue-equivalent" material is added as part of the dosimeter. For Co^{60}, equilibrium was obtained with 400 mg/cm^2 of material. Under these conditions the energy absorbed in the emulsion results primarily from the secondaries generated in the surrounding medium and can readily be determined.

EHRLICH and FITCH[10] were the first to design and calibrate a film badge employing this principle. As a consequence all present Co^{60} γ-ray exposures were made in a modified version of their badge. The outer filter was a combination of 1.1 mm of tin and 0.3 mm of lead while the Bakelite container housing the film was reduced in thickness to 5 mm. It is of interest to note that the average sensitivity of the three various

Table 1. *β-Ray exposure rates*

Source	Distance from source	Absorber (mg/cm^2)	e.s.u./cm^3 per hr	rads/hr
U disk	Contact	7	0.240	0.228
U disk	Contact	80	0.150	0.1425
Sr^{90} disk	Contact	107	16.2	15.6
Sr^{90} point source	15.8 cm	$\approx$100	29.7	28.2

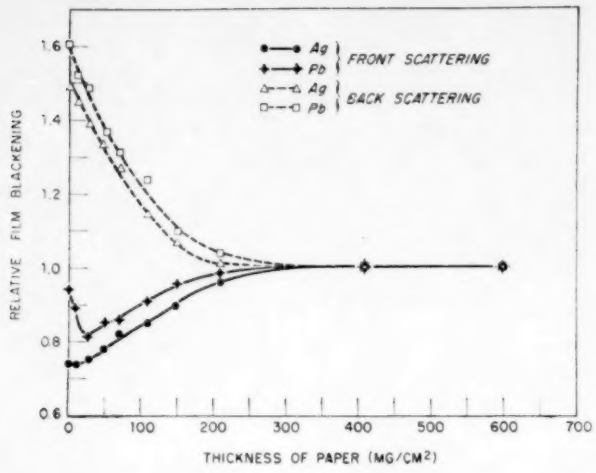

FIG. 2. Effect of forward- and back-scattered electrons from Ag and Pb on film.

films exposed in badges using bare Ag and Pb absorbers was, respectively, 9 per cent less and 9 per cent greater than in the EHRLICH–FITCH dosimeter.

Films investigated included Eastman Type K and DuPont Type 555 dosimeter films and Eastman NTA nuclear emulsion. Only the Type K and NTA films were single-coated emulsions. For the double-coated films the β-ray dose to the center of the emulsion rather than to the front surface was used. All β-ray exposures were made with bare film backed with black paper and plastic.

Complete characteristic curves were obtained for each source. It had previously been established that characteristic curves identical in shape will be obtained for β-rays and Co^{60} γ-rays.[11] Fig. 3 illustrates this condition for DuPont Type 502 film. The data are normalized so that one curve can be drawn through both sets of points.

In Table 2 the relative sensitivities of the different films (based on horizontal displacement of the characteristic curves) are tabulated for the different β- and γ-ray exposures. A value of 1.0 is arbitrarily assigned the uranium exposures with 7 mg/cm^2 absorber. The data for uranium exposures with 80

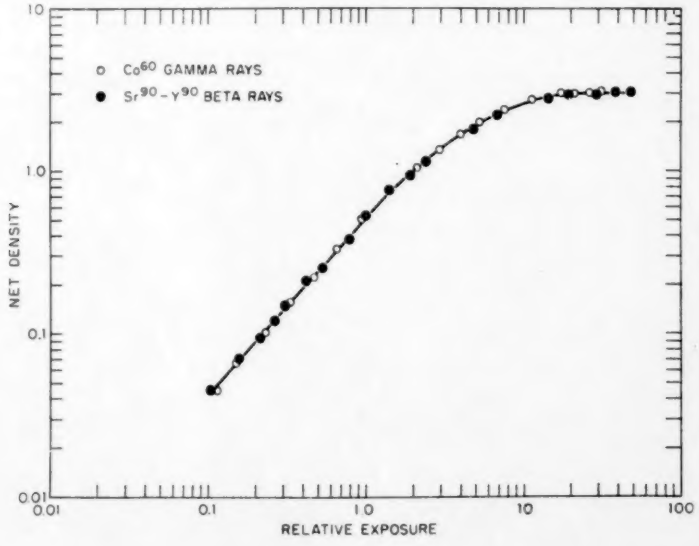

FIG. 3. Type 502 film characteristic curve obtained with relative β-ray and Co^{60} γ-ray exposures.

mg/cm² absorber agreed within experimental error and can be considered identical to the given values for 7 mg/cm² absorber.

Types 555 and K films are 3 per cent more sensitive to the Sr^{90} disk source than to uranium. Comparisons of identical Sr^{90} exposures with a disk and point source show agreement within 1 per cent and within 7 per cent for the respective films. On the other hand, the relative film sensitivity to β-rays averaged 11 per cent greater than for an equivalent γ-ray exposure with Co^{60}. These differences in β- to γ-ray sensitivity are not apparent for the Eastman NTA nuclear emulsion.

Table 2. *Relative film sensitivities to uranium and* Sr^{90} *β-rays and* Co^{60} *γ-rays*

Film	Uranium disk	Sr^{90} disk	Sr^{90} point source (15.8 cm)	Co^{60}
DuPont 555	1.0	1.03	1.02	0.91
Eastman K	1.0	1.03	0.96	0.87
Eastman NTA	—	0.98	1.0	1.0

DISCUSSION

Studies of film response to charged particles indicate that conventional dosimeter films are comparable to electron sensitive nuclear emulsions. The average grain sensitivity is such that any singly charged particle passing through a grain will impart sufficient energy to produce a latent image. This is further confirmed by the fact that phenomena normally associated with the multiple-hit picture, namely reciprocity law failure and intermittency effects, are not observed with exposures to ionizing radiations.

Fig. 4 shows the dosage sensitivity of Eastman Translite dosimeter film and three nuclear emulsions as a function of the rate of energy loss and also as a function of electron and proton energies.[4] A sensitivity of 100 per cent is assigned to particles of minimum specific ionization. The solid lines are dosage sensitivity curves obtained by the authors with direct density measurements. The dashed curves are measurements of grain density obtained by Brown *et al.*[12] in nuclear track plates. Their data were recalculated to conform with the current presentation. It is of importance to note that the dosage sensitivity of Translite (and other conventional dosimeter films) and Kodak NT-4 electron sensitive emulsions decreases rapidly with decreasing particle energy.

The electron energy distribution produced by the interaction of γ- rays has been independently calculated by Grey[13] and by Cormack and Johns[14]. Table 3, taken from Grey[13], gives the mean linear ion density per micron of tissue for different β-, X- and γ-radiations. Column 4 gives the predicted dosage sensitivity of dosimeter film as determined from the data in column 3 and the curve for Translite film in Fig. 4. The data indicate that a 5 per cent greater film sensitivity would be expected for β-rays than for Co^{60} or radium.

Cormack and Johns[14] pointed out the difficulties that arise with simple calculations in trying to account for all the effects of scattered radiation for X-rays interacting with matter. In order to obtain a more valid representation of the secondary electron distribution from Co^{60}, direct experimental measurements were made in a water phantom with an organic scintillation spectrometer.[15] From their data of energy distribution for a 5 × 5 cm field at a 2 cm depth in water, the rate of energy loss was calculated to be 2.4 times that of minimum. From the β-ray spectra of Fig. 1, a similar calculation determined the effective rate of energy loss to be 1.6 times that for a theoretical minimum. The resultant dosage sensitivity for the γ- and β-ray sources was then evaluated as 87 and 95 per cent, respectively. This value is in reasonable agreement with the relative uranium β-ray to Co^{60} γ-ray sensitivities for dosimeter film observed for the respective sources given in Table 2.

It is further noted that the data in Table 2 for NTA nuclear track plate film shows a difference no greater than 2 per cent between the two types of radiation. This again is consistent with the curves of Fig. 4 if reference is now made to Eastman NTB and Ilford C-2 proton sensitive emulsions (both films are comparable to NTA). The ability of these emulsions to measure dose

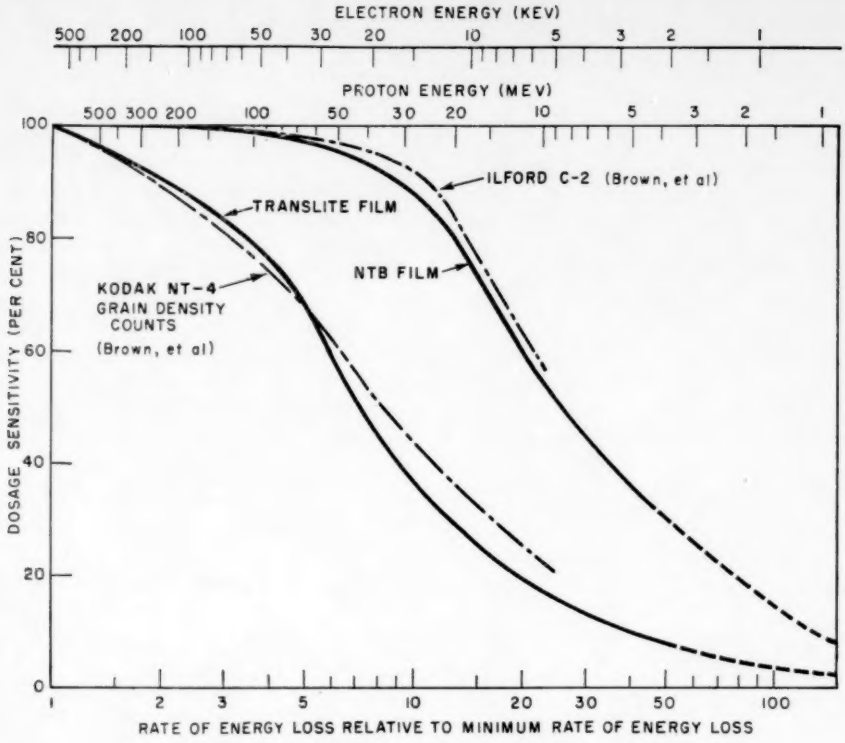

FIG. 4. Dosage sensitivity of nuclear and photographic emulsions.

Table 3. Ion densities produced by different ionizing particles

Radiation	Mode of generation	Mean linear ion density (ions per μ of tissue)	Dosage sensitivity of film to the ionizing electrons (%)
Very high energy β- and γ-radiation	Theoretical minimum ion density for any particle	6.3	100
	20–30 MeV X-rays, natural and artificial radio elements	8.5	97
γ-radiation	Co^{60} or radium	11	92
X-radiation	"Supervoltage" 1000 kV	15	88
	"Deep therapy" 200 kV	80	31
	Diagnostic 30–180 kV	100	25

relatively independent of energy was explained by WEBB[16] on the basis that particles of minimum specific ionization will require many interactions to produce a latent image. Up to the point where a single particle can release enough energy within a grain to produce a latent image, film blackening will be proportional to dose. Beyond this point the emulsion assumes the properties of a particle detector and the shape of the dosage sensitivity curve parallels that for dosimeter films.

SUMMARY

Within 3 per cent identical sensitivities were observed for dosimeter films exposed to a uranium β-ray standard and Sr^{90} when both sources were filtered to produce similar β-ray spectra. Differences in exposure geometry between an extended and point source of Sr^{90} also had minimum effect on film response.

Relative film sensitivity decreased by approximately 10 per cent when the emulsions were exposed to an equivalent amount of ionizing radiation from Co^{60} γ-rays. Similar measurements made with Eastman NTA nuclear emulsions showed no difference in film response for β- or γ-radiation.

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THE LATENT PERIOD AND ITS VARIATION IN HUMAN LEUKAEMIA INDUCED BY X-RAYS

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Abstract—The distribution of time intervals between irradiation by X-rays and onset of acute or chronic myeloid leukaemia has been investigated, in particular for the ankylosing spondylitics in COURT-BROWN and DOLL's study series. Two methods of analysis are given in outline, both of which allow (a) for several courses of irradiation on one patient, any of which might have induced leukaemia, and (b) for limited information on longer time intervals because there were many recent treatments. Intervals in sub-groups, e.g. for acute leukaemia only, and for "doubtful" cases only, are also given separately.

The probability of onset is found to increase suddenly to a maximum in the fourth year after the inducing irradiation. The corresponding maximum incidence among all 3-year-old children is discussed in relation to the time intervals following irradiation of a fœtus or a young baby.

The analysis in general confirms that leukaemia can be caused by X-rays but throws doubt on the assumed linear dose-response relationships, which depend upon assuming a uniform distribution of latent periods. Accepted ideas on risk in terms of dose to the spinal marrow, and on mechanisms of induction, are briefly criticized.

1. INTRODUCTION

IN ASSESSING human radiation hazards, direct evidence from case records is of enormous value. Very precise information needs to be extracted from such records, which therefore merit being analysed very fully. This paper gives the outline of a detailed analysis of a well-known study series. The series consists of patients heavily irradiated by X-rays who afterwards developed leukaemia. Neither the study nor the analysis is complete, but it seems worth while to publish some of the results now. They are already interesting enough, and they can help with some current controversies in this urgent problem of the incidence of leukaemia following ionizing radiation.

It is now well established that the risk of developing leukaemia is increased following heavy doses of ionizing radiation. This radiation can consist, for example, of therapeutic X-rays.[1–3] FABER[4–6] finds increased incidence in patients compared with controls, not only after therapeutic irradiation but also (to a much lesser extent) after diagnostic irradiations. STEWART *et al.*[7] find, for malignancies in children (about half of which were leukaemias), about twice as many mothers who had been irradiated while pregnant as irradiated controls. Even among patients who have been heavily irradiated the proportion developing leukaemia is very small, but it is much larger than the proportion for the whole population. However the number of people exposed to small doses of radiation amounts to many millions, so that it is urgent to find out as exactly as possible how the risk (of getting the disease) is related to the dose, in other words, the shape of the dose–response curve for all doses, including very small ones.

One of the objects of studying latent periods is then to obtain more information from human case records and to use it to calculate dose–response curves much more accurately. The basic assumption is that there is a small risk of developing leukaemia induced by external ionizing radiation (in this paper the only radiation considered is X-rays) and that this risk to any

one patient changes in time. For cases of leukaemia following heavy single doses of X-rays (or courses of treatment extending over only a few months), the time intervals between dose and onset do in fact vary widely, in particular there are many intervals between 3 and 8 years.

In all other kinds of cancer there is strong evidence for much longer latent periods, of the order of decades. However, it is still not certain whether there is a risk of getting leukaemia for only a few years, in which case, although serious, this risk may be judged differently from a nearly permanent danger.

2. THE CASES OF LEUKAEMIA IN ANKYLOSING SPONDYLITICS TREATED BY X-RAYS

The most extensive data on doses up to now have been obtained in the survey directed by COURT-BROWN and DOLL[3]. It has the great advantage of being a survey of a defined population. This consists of 13,000 ankylosing spondylitics treated by X-rays. Up to the end of 1955 at least thirty-two and probably thirty-seven of these patients had developed leukaemia; eight more cases have so far been confirmed in 1956, 1957 and 1958. They estimated that the number of cases, up to the end of 1955, would have been about three if the leukaemia occurred independently of irradiation. They quote independent evidence[8] that ankylosing spondylitis is not associated with leukaemia in unirradiated patients. (Later ABBATT and LEA[9] suggested that the diseases may be slightly associated, but this would not upset the conclusions reached in the present paper. In fact the new analysis of latent periods provides more evidence that most cases were associated with previous irradiations, although possibly not all of them.)

The dates, sites and amounts of the doses were recorded and/or calculated in detail. This was done for all these cases, for many more that COURT-BROWN and DOLL classed as "doubtful" cases in which leukaemia was suspected (but aplastic anaemia, for example was afterwards preferred as a diagnosis), and for a large sample, about 2000, of the rest of the spondylitics. This sample was chosen to yield more precise estimates of doses to heavily treated patients. The doses to the spinal marrow, and also the whole-body integral doses, were estimated for each course of treatment using detailed case records.

All this was extremely laborious. The main findings are so important that they were rightly published by COURT-BROWN and DOLL as soon as possible, although only meant to be provisional. Briefly, if the dose to the spinal marrow is the appropriate measure of risk to the patient, incidence of leukaemia seems roughly proportional to the total dose up to 12 months before onset. If the measure of risk is the whole-body integral dose the incidence is roughly proportional to the square of the dose. Both relationships imply that there is no threshold dose below which the increased risk is negligible.

3. THE IMPORTANCE OF THE LATENT PERIOD

These findings depend on the assumption that the distribution of latent periods is zero up to 12 months after irradiation and uniform after that. In other words the probability of developing leukaemia in any one year is assumed to be proportional to the total dose up to 12 months before the middle of that year. If this distribution is not uniform the risk of developing leukaemia depends upon the time that has elapsed since the doses were given. The dose–response relationship for a very uneven distribution of latent periods was worked out by COURT-BROWN and DOLL. It is very irregular, but still shows an increase of incidence with dose (see Table 10).

It was not clear that the latent periods could be estimated in all cases, for most patients had several courses of treatment, each of which could contribute to increasing the risk of getting leukaemia. But this can be done, and this paper gives the preliminary results of this fuller analysis. The second major difficulty arises because there were many recent irradiations. There is very little information available, for example, for time intervals beyond 8 years in the first study. This study included leukaemias only if diagnosed before the end of 1955, and most irradiations took place less than 8 years before this. The final follow-up study will include 4 more years, and hence more evidence

relating to the time intervals up to 12 years. The methods of analysis seem to be new; they may be of more general interest to medical statisticians than to health physicists and are best given in detail elsewhere, but the underlying principles will be given in this paper, together with the main results. The new findings are not conclusive, but they provide strong evidence that the distribution of latent periods is far from uniform and hence that the dose–response relationships may be different from those published so far.

4. ESTIMATING THE DISTRIBUTION OF LATENT PERIODS FROM CALCULATIONS OF THE PROBABILITY OF ONSET IN A GIVEN YEAR

In estimating incidence, obviously the population at risk must be known, and detailed information on the doses to patients who escaped leukaemia is as essential as that on the cases. But for estimating latent periods, most of the relevant information comes from the actual cases of leukaemia; in fact it is not only simpler but much more accurate to use data from these cases only. If this is done, the estimates then depend very little upon any assumed relation between dose and risk (of developing the disease); it must be assumed, of course, that the risk increases with the dose. For example, one patient might have had two-thirds of his total dose given $3\frac{1}{2}$ years before the onset of leukaemia, and the remaining third 6 years before the onset. This record then provides evidence for latent periods of $3\frac{1}{2}$ or 6 years. There may be errors in assessing the relative effects of these two treatments, which would change the weighting given to the estimated probabilities of onset after $3\frac{1}{2}$ and after 6 years, but these errors cannot change the time intervals. Also, other things being equal, the risk probably increases with the age of treatment for an adult (see, for example, HEWITT[10]). This again, can be disregarded without much error when we work with relative probabilities.

These probabilities are defined in the first place in terms of a single dose (or a series of irradiations) given over a short period. Given that leukaemia did develop and was caused by the irradiation, we write p_k for the probability that it developed in the kth year, that is, between $k - \frac{1}{2}$ and $k + \frac{1}{2}$ years from the middle of the period of irradiation. It is also assumed that the *relative* values of the probabilities (the various values of p_k) do not depend on the size of the dose, and that their absolute values are small. (Provided they are small enough, contributions to the risk due to the various probabilities can be added linearly in the mathematical analysis.) For this problem the main interest is in the relative probabilities.

In Table 1 the doses in any one year are given as percentages of the total dose to the patient. In case (1), for example, diagnosed in 1953, 45.4 per cent of a total dose to the spinal marrow of 471 r was given 13 years before 1953, that is in 1940, and 54.6 per cent of 471 r in the year 1953 minus 12, i.e. in 1941. For this first analysis the total dose in any one year is regarded as being given in the middle of that year. Then the numbers below the tabulated percentages give the intervals to the nearest year between irradiation and diagnosis.

The cases were included in the series, provided that leukaemia was diagnosed before the end of 1955. Case (12), for example, could have been so diagnosed in any year up to the tenth after the first dose. Given that leukaemia did develop during this period. the relative probabilities that this happened during 1946, 1947. . . up to 1955 can be worked out. They are quite complex. In the years after 1948, when the third treatment was given, each course of treatment contributes to this risk. If, in comparing contributions from different doses, the increased risk is proportional to the dose, the relative probability of onset in 1950 will be:

61.7 times the probability of onset in the second year after irradiation
plus 9.6 times the probability of onset in the fourth year after irradiation
plus 28.8 times the probability of onset in the fifth year after irradiation

divided by the overall probability, obtained by summing expressions like this for each year from 1946 to 1955. It is because of this division that only the relative values of the doses come into the calculation, and not the absolute values of either doses or probabilities.

These "conditional probabilities" can be

Table 1. *Summary of the basic data, adapted from* COURT-BROWN *and* DOLL's *Table* 17, *to show intervals between irradiation and diagnosis of leukaemia*

Case no.	Age at first treatment	Diagnosis in year	Total dose in spinal marrow (*r*)	Doses in % of total dose No. of years between dose and onset					
1	22	1953	471	45.4 13		54.6 12			
2	44	1948	3535	9.4 8	17.5 7	17.9 6	10.3 5	33.4 4	11.4 1
3	32	1950	2622	34.7 8	21.2 7	14.2 6	26.0 5	3.8 4	
4	31	1947	2706	7.2 5	54.9 4	15.9 3	22.0 2		
5	37	1955	1395	7.3 13	58.3 5	5.5 4	17.8 3	5.5 2	5.5 1
6	23	1950	2466	15.0 7	44.0 6	41.0 2			
7	31	1953	1303	39.9 9	2.4 8	17.0 5	13.6 4	27.2 3	
8	22	1955	1559	60.4 11	20.3 10	19.4 1			
9	37	1948	1059	71.4 4	28.5 2				
10	36	1955	112	38.4 10	61.6 4				
11	35	1952	1763	53.7 7	23.8 6	13.1 4	9.4 1		
12	50	1950	1182	28.8 5	9.6 4	61.7 2			
13	58	1951	746	100 5					
14	27	1954	1130	36.5 8	63.5 6				
16	42	1952	1849	42.9 6	40.3 3	16.8 1			
17	45	1955	2304	24.0 8	36.6 7	1.5 6	9.6 5	24.6 4	3.7 2
18	31	1954	1802	57.8 7	42.0 1				
19	63	1952	1016	100 5					
20	26	1954	656	51.8 7	48.2 1				
21	47	1951	967	31.5 4	42.6 3	2.7 2	23.2 1		
22	26	1954	2195	49.9 6	50.1 1				
23	30	1952	1654	97.2 4	2.8 3				

Table 1—*(contd.)*

Case no.	Age at first treatment	Diagnosis in year	Total dose in spinal marrow (r)	Doses in % of total dose No. of years between dose and onset		
24	54	1953	1016	100.0 5		
25	27	1952	609	100.0 5		
26	50	1954	2295	17.3 6	39.6 4	43.1 3
27	37	1955	903	43.0 6	57.0 5	
28	44	1953	721	100.0 4		
29	65	1954	727	100.0 4		
30	53	1954	644	100.0 4		
31	26	1952	931	33.8 2	66.2 1	
32	69	1953	840	100.0 2		
35	26	1955	1053	100.0 2		
37	42	1953	2328	74.1 11	25.9 3	
38	53	1952	901	86.2 4	13.8 2	
39	40	1953	470	66.6 4	33.4 3	
40	49	1953	1390	75.1 4	24.9 3	
41	70	1955	1045	100.0 4		

written down mathematically for each case, then a distribution of latent periods, in fact of p_k values, can be calculated that yields the maximum probability of obtaining the observed distribution of years of onset. The dates of diagnosis as given by COURT-BROWN and DOLL were used: exactly the same analysis can be performed using dates of death. No information is available beyond the thirteenth year, and very little for periods beyond 8 years, because nearly all treatments took place less than 8 years before the end of 1955. The follow-up survey will, of course, include at least 3 more years and so more information on these longer periods. (Since this information will be available fairly soon, I have not tried to obtain an optimum solution, only one somewhere near it.) A distribution fairly near this optimum is given in Table 2. The values of p_k are, as already defined, the probabilities for onset in the kth year following the dose, that is between $k - \frac{1}{2}$ and $k + \frac{1}{2}$ years afterwards, but subject to the condition that one dose can cause leukaemia only up to $13\frac{1}{2}$ years afterwards (i.e. $p_1 + p_2 + \ldots + p_{13} = 1$). The sharp maximum during the fourth year is what we would expect

Table 2

k (years)	1	2	3	4	5	6	7	8	9	10	11	12	13
$100\,p_k$	0	7.0	0.6	24.7	10.9	6.4	6.4	6.4	7.5	7.5	7.5	7.5	7.5

after looking at Table 1. The secondary maximum at 2 years may be due to chance.

The optimal distribution can be determined, in principle, in the following way. Suppose doses D_1, D_2 and D_3 are given in the years $Y - y_1$, $Y - y_2$ and $Y - y_3$, respectively. Records are available up to the end of year Y. The unconditional probability that one of these doses induced leukaemia diagnosed during or before this year is assumed to be:

$$G = D_1P(y_1) + D_2P(y_2) + D_3P(y_3)$$

where $P(y_s) = p_1 + p_2 + \ldots p_{y_s}$, $s = 1$, 2 or 3 i.e. $P(y_s)$ is the unconditional probability of onset, diagnosed in the first y_s years following the D_s irradiation.

The year of diagnosis is denoted by $Y - l$. The conditional probability for this year, given that leukaemia developed before the end of year Y, is then

$$\frac{g}{G} = \frac{D_1p(y_1 - l) + D_2p(y_2 - l) + D_3p(y_3 - l)}{D_1P(y_1) + D_2P(y_2) + D_3P(y_3)}$$

where the functional notation $p(y_k - l)$ has been used instead of p_k; $k = y_k - l$.

A similar ratio g/G is formed corresponding to each individual dose record. The number of terms in g and in G is obviously equal to the number of years during which the patient was irradiated. The overall conditional probability is the product of the g/G, or the sum:

$\Sigma(\log g - \log G)$; it has a maximum, for variations in $p_1, p_2 \ldots p_{13}$ $(\sum_1^{13} p_i = 1)$, when

$$\sum_{\text{all cases}} \frac{\partial}{\partial p_i} (\log g - \log G) = 0, \; i = 1 \text{ to } 13$$

Only the ratios of the p_i can be determined so that twelve of these equations suffice. The coefficients D are proportional to the doses, defined in one way or another.

In a trial solution, the G can be estimated more accurately than the g. There was so little information available for the longer intervals that the corresponding p_i were assumed equal. In improving on a trial solution, the p_i multiplying the smallest D's in any one function g are regarded as known, hence, in each partial derivative $\partial g/\partial p$ the number of unknowns is only one or two. Despite this, the calculation is lengthy and the optimal solution changes (as it should) when the final year Y changes.

If the distribution in Table 2 is the true one, the observed joint distribution of the years of diagnosis, for all thirty-seven cases, is found to be 3225 times more probable than it would be for a constant distribution of latent periods after 1 year, $(p_0 = 0,\ p_1 = p_2 = \ldots = p_{13})$! Even with this large ratio, most of the treatment records do not obviously favour one distribution more than the other. The (geometric) mean ratio of probabilities is only 1.24 to 1 ($3225^{1/37} \simeq 1.24$), so that little or no difference is noticed on inspecting the records of cases having many treatments. And this high ratio does not signify nearly as much as a 1 in 3000 significance level would, since changing six constants (values of p_k), in order to fit the data better, increases this ratio anyway. But the fact that the distribution of Table 2 is fairly regular is important. The sharp rise between the third and fourth year is noteworthy; in fact no case developed leukaemia in the third year after his largest dose, but many did so during the fourth year.

This method has some disadvantages. The optimum values of some p's can be negative! Also it is difficult to devise a "goodness of fit" test comparing observed with expected times of onset. Further, if onset had occurred, say, between the first and second doses, only the first would have been recorded. To meet these difficulties a second method has been tried.

5. COMPARISON OF OBSERVED AND EXPECTED DISTRIBUTIONS OF LATENT PERIOD BY MEANS OF "PERCENTAGE SCORES"

The first analysis, above, was based on doses to the spinal marrow; the second will use whole-body integral doses. In this way we

can directly verify that the best-fitting distribution depends only slightly upon which measure is chosen. The complete case records were used with dates of treatment and of diagnosis given to the nearest month (COURT-BROWN and DOLL[3], Appendix A). When one course of treatment extended over several months, it was assumed to take place during (the nearest month to) the middle of this period. Table 3 gives dates of onset and of death, in months from December 1939, and, again, the intervals between doses and diagnoses of leukaemia. Two cases (nos. 8 and 28) were left out because complete records were not available at the time; also one case (no. 10) of chronic lymphatic leukaemia. According to FABER's[6] recent extensive studies, the incidence of this type of leukaemia is no higher among previously irradiated patients than among non-irradiated patients. Also, following FABER[4–6], the three cases of chronic myeloid leukaemia are given separately. This leaves thirty-one acute leukaemias.

The observed distributions of latent periods is now obtained from a simple scoring system. In case (2), for example, 17.1 per cent of the whole-body integral dose was given 13 months before diagnosis; this is scored in the second interval, which extends from 12 to 23 months (strictly this included all intervals between $11\frac{1}{2}$ months and $23\frac{1}{2}$ months). In the fifth interval, 48 to 59 months, is scored $17.3 + 7.8 + 14.8 = 39.9$; in the seventh interval, 72 to 83 months, $13.3 + 14.1 = 27.4$, and in the interval, 84 to 95 months, 15.7. There is a score of 1 allocated to each individual record, and distributed over the various intervals. In Table 4, row (b) gives the observed distribution of scores obtained in this way.

The distribution of scores is obviously related to the latent period distribution, but obtaining one exactly from the other necessitates a complex calculation. The observed distribution is heavily biased towards short intervals. This is mainly for two reasons, already mentioned. The first is that doses given less than, say, 60 months before the end of 1955 cannot contribute any scores to intervals greater than 60 months. The second bias is less obvious and operates only when there is more than one course of treatment. Suppose a patient is irradiated by equal doses in 1948 and 1951. If leukaemia was diagnosed in 1953 scores of 50 per cent would be allocated to a 2-year and to a 5-year interval. But leukaemia could have developed before 1951, induced by the first dose, and then 100 per cent would have been scored to an interval less than 3 years. The bias due to this effect obviously increases with the interval between the first and last periods of irradiation.

In the calculations summarized in Table 4 both biases have been allowed for.

There is even a third bias, which may be small. Suppose all patients are irradiated every 3 years. Then when estimating the *expected* score due to onset of leukaemia before the end of 1955 (or whichever the final year is) we would have to allow for this third irradiation, even if leukaemia developed before 1954. This would increase the expected scores for the smaller intervals. In fact treatments tend to become less frequent; moreover their frequency decreased sharply after 1955, when the first results of the study became known. Hence this bias may become small when the final analysis is made, up to the end of 1959. An approximate correction will then be made; it is neglected in the analysis given here.

The expected scores β_1 and β_2 in Table 4 arise from two different assumptions about the distribution of latent periods. In both cases it is assumed that leukaemia was diagnosed not more than 14 years after the first treatments and before the end of 1955, whichever is the earlier. Each dose is expressed as a fraction of the total, as in Table 3. The probability of onset due to the ith dose, $100D_i$ per cent of the total dose to any one patient, is assumed to be

$$D_i p_j$$

for the period of 12 months from j years after this dose, that is for the 12 months between $12j$ and $12j + 11$. This is the most natural definition for p_j when the unit of time is a month instead of a year, although it does not correspond to the one in Table 1.

Suppose onset is in the jth year, and that this year comes after all the courses of treatment. Then each treatment will contribute scores $D_1, D_2, D_3, \ldots$ to the various intervals between irradiation and onset. For onset before the last treatment D_m, say, the remaining doses must be expressed as proportions, not of unity, but of $1 - D_m$.

Table 3. *Similar data to Table* 1, *but using exact dates of treatment and of onset* (COURT-BROWN and DOLL[3], Appendix A)

Case no.	No. of Months from Dec. 1939		Acute leukaemias
	Month of diagnosis Month of death	Total dose whole-body integral (mg r)	Doses as % of total Intervals in months between middle of course of treatment and diagnosis of leukaemia
1	163 167	12.2	58.2 7.4 34.4 156 148 146
2	100 104	103.8	15.7 14.1 13.3 14.8 7.8 17.3 17.1 88 76 73 56 55 49 13
3	127 134	76.1	9.9 6.2 4.5 1.7 12.4 1.3 4.7 10.4 5.3 13.0 13.8 6.8 3.2 7.0 99 96 93 91 89 86 83 78 67 61 58 52 51 49
4	87 95	65.5	11.8 24.6 16.8 5.2 12.1 2.6 11.6 7.9 7.5 58 49 44 32 23 15 10 6 3
5	182 184	50.7	6.5 29.6 13.4 29.2 10.7 10.7 146 49 38 36 20 8
7	157 157	39.2	38.8 3.8 26.8 1.5 5.4 7.9 15.8 106 105 86 84 52 41 33
9	107 107	13.2	80.3 19.7 48 24
11	156 158	27.8	28.4 28.4 15.1 20.5 7.6 92 88 80 57 13
12	127 130	10.5	20.0 7.6 72.4 56 53 21
13	134 135	15.0	100.0 57
14	177 178	26.4	10.6 11.0 11.0 67.4 100 97 94 71
16	145 149	31.8	48.4 37.7 13.8 65 27 1
17	185 188	51.9	4.6 6.0 6.0 4.8 7.3 30.3 9.1 3.7 2.7 11.8 10.4 3.5 100 98 95 90 83 78 68 62 50 45 41 17
18	169 173	31.2	57.4 42.6 82 8

Table 3—(contd.)

Case no.	No. of months from Dec. 1939		Acute leukaemias
	Month of diagnosis Month of death	Total dose whole-body integral (mg r)	Doses as % of total Intervals in months between middle of course of treatment and diagnosis of leukaemia
20	179 179	9.1	4.4 39.6 53.8 2.2 84 83 19 9
21	142 142	26.3	21.7 25.9 17.1 13.7 21.7 47 45 36 27 18
22	177 177	36.0	7.8 12.2 30.8 49.2 79 76 72 62
23	150 151	44.0	87.3 12.7 50 37
24	161 181	16.5	100 58
25	149 153	9.3	100 46
26	177 178	53.1	13.0 31.1 49.0 7.0 73 46 42 36
27	189 192	18.8	46.3 53.7 71 58
29	179 181	24.4	100 50
31	147 147	18.2	31.3 34.1 34.6 14 11 5
32	162 165	15.3	100 25
35	186 186	18.1	100 28
37	160 161	60.9	37.8 38.3 21.2 2.8 134 130 31 8
38	152 152	18.3	70.5 29.5 46 30

Table 3—(contd.)

Case no.	No. of months from Dec. 1939		Acute leukaemias
	Month of diagnosis Month of death	Total dose Whole body integral (mg r)	Doses as % of total Intervals in months between middle of course of treatment and diagnosis of leukaemia
39	159	30.0	33.3 33.7 33.0
	164		47 44 36
40	167	29.4	70.4 29.6
	168		50 36
41	186	17.8	100
	187		42
Chronic myeloid Leukaemias			
6	121	56.3	22.6 23.4 25.6 28.4
	144		76 74 68 15
19	150	14.6	100
	151		59
30	174	12.0	100
	174		44

Suppose the second and third treatments are m_2 and m_3 months after the first. The first treatment will contribute $D_1^2 p_j$ to the expected scores for each interval $12j$ to $12j + 11$. It will contribute $D_1 D_2 p_j$ to the expected score for each of the intervals $12j - m_2$ to $12j + 11 - m_2$ months; similarly $D_1 D_3 p_j$ to the interval from $12j - m_3$ to $12j + 11 - m_3$.

Clearly the expected scores for each interval involve terms corresponding to every product $D_i D_j$ and the calculations increase in length very quickly with the numbers of treatments to one patient. In some cases treatments given within a few months of one another were assumed to be given during one month. This was only done in calculating expected scores, never when summing observed scores.

In calculating the values in row (β_1) all the values of p_j are assumed to be equal (with no zero for the first 11 months). In row (β_2) the p_j are assumed to form a step function, i.e. to have the constant values of row (a) for each month in an interval covering 12 months, (i.e. a time interval between $12j$ and $12j + 11$ for $j = 0, 1, 2, \ldots, 14$).

It is clear that the expected scores are much nearer the observed ones when the true distribution of latent periods is (a) than when it is uniform (all p_j equal). The same holds good for the three chronic myeloid leukaemias (see Table 7). On the evidence from thirty-one acute leukaemias, the probability of onset is greater in the third and fourth years after inducing irradiation, and is highest in the fourth year, that is, between 48 and 59 months. In the meantime fifteen more cases of acute leukaemia in the series have been confirmed. The details of doses are not yet available for all of them; also they cannot be included in this accurate analysis until the defined period is extended to the end of 1959. Then it is hoped

Table 4. *Sums of percentage scores for intervals between middle of course of treatment and diagnosis of leukaemia, both given to nearest (middle of the) month*

In each case the score of 100 is distributed among the intervals according to the % of total dose (in whole body mg r) taking place within that interval.

(a) Trial distribution of latent periods: values of $100p$ when onset is less than 14 years from irradiation. (p is then the probability of onset in the given range of time intervals. Since times are measured in months, the ranges of intervals have not been made to correspond to those in Table 2.)

(b) Observed sum of scores for thirty-one acute leukaemias.

(β_1) Expected sum of scores for thirty-one acute leukaemias, corresponding to a uniform distribution of latent periods between 0 and 167 months.

(β_2) Expected sum of scores when the probabilities of diagnosis in given intervals of time are given by the tabulated values in percentages in row (a). [(β_2'): the same with distribution of Table 5 in place of (a).]

	Intervals (months)														
	0–11	12–23	24–35	36–47	45–89	60–71	72–83	84–95	96–107	108–119	120–131	132–143	144–155	156–167	Total
(a)	0	2	4	15	15	8	7	7	7	7	7	7	7	7	100
(b)	167.8	292.0	342.8	649.1	784.6	193.2	260.4	142.5	90.9	0	38.3	37.8	48.3	58.2	3095.9
(β_1)	475	470	455	413	355	303	207	129	98	78	44	33	25	16	3101
(β_2)	106.0	275.5	443.1	800.7	667.4	287.7	179.9	108.6	80.4	60.2	34.9	24.3	17.4	10.9	3097
(β_2')	87	218	355	681	825	346	205	134	89	61	38	28	20	12	3099

(b), (β_1), (β_2), (β_2') } 3100

Table 5. *A possible distribution of latent periods. Suggested estimates of p based on the results to date: values >83 months very unreliable*

Intervals (months)	0–11	12–23	24–35	36–47	48–59	60–71	(8 years) 72–167
$100\,p$	0.2	1.8	3.5	14.5	23.0	9.0	48.0 (=8 × 6% per year)

that all cases will have been included from the 13,000 spondylitics. The combined evidence so far suggests that the distribution may be more sharply peaked than the estimate in Table 4, and that the maximum risk is at about 50 months after irradiation. Table 5 gives a provisional estimate.*

As yet there is little evidence relating to the long latent periods; again, more will be available in the final analysis. At present it seems that the risk decreases only slowly between the eight and fourteenth year; however the total score for intervals greater than 96 months is equivalent to about nine complete cases only. FABER's evidence (see Section 7) and that from Hiroshima following γ-radiation, (COBB *et al.*[12] or BURCH[11]) suggest that the risk falls off fairly quickly.

The most definite conclusion is that the probability of onset is very low during the first 3 years after the inducing radiation. This is in conflict with the only published estimate of

* A second calculation has now been made on the same data to test this estimate; it appears to be a good one, at least for intervals up to 6 years [see row (β_2') in Table 4].

latent-period distributions (COBB *et al.*[12], Fig. 2). The latter is based on the forty-one leukaemias confirmed by COURT-BROWN and DOLL together with ten American case records. They do not state how far back in time these records extend; if they are recent records the same kind of bias would operate against longer intervals; the authors suggest this but do not try to allow for it. In dealing with multiple doses they replace these by a single dose given at a date that is a weighted average of the actual dates of irradiations. Hence a dose of 200 r in the hundredth month (zero = December 1939), followed by 100 r in month 130, was presumably regarded as a single dose given during the 110th month. It seems impossible to justify this when estimating a *distribution* of latent periods; even when estimating a mean latent period the arguments above show that these are biased, and Table 3 shows that the bias is quite large. They give a table showing that estimates of the *median* latent period are similar for short and long intervals between first and last courses of treatment, but all estimates will be biased; the causes of bias will differ according to the number of treatments, as we have seen; the amount of bias might well be about the same. However these estimates do suggest a maximum near the forty-eighth month, as does COURT-BROWN and DOLL's discussion of the cases receiving only one course of treatment.

In the rest of the paper, I shall discuss the latent period for leukaemia in children, the Danish survey, the effect on dose response curves, and possible mechanisms of leukaemogenesis. First we shall see how the analysis of time intervals can be used to test "doubtful" cases.

6. COMPARISON OF "DOUBTFUL CASES" WITH THE REST

COURT-BROWN and DOLL classed five cases (nos. 37 to 41) as "doubtful"; for in these cases leukaemia was strongly suspected but not certain. In five more cases (nos. 2, 3, 27, 29 and 32) leukaemia was not so certified at death: also, the certified cause of death for case (1) was chronic lymphatic leukaemia, whilst the "preferred diagnosis" was acute myeloid leukaemia. BRUES[13] calls the whole study "painstaking" and one of his grounds for questioning the main findings is the "high proportion of equivocal cases". COURT-BROWN and DOLL however, classify cases as definite or doubtful according to the post-mortem findings, which establish the condition much more clearly than that of an average case with only a death certificate diagnosis. But it is interesting and worth while to apply an independent statistical test for the "doubtful" cases, including for this purpose the cases with a different cause of death on the certificate, and to find the distribution of latent periods for these eleven cases separately. Case (1) is classed as a "doubtful" one in view of FABER's evidence. Table 6 is then calculated in the same way as Table 4, but only for these cases.

If onset is independent of irradiation, the distribution of intervals would be approximately uniform if no patient ever had more than a single course of therapeutic X-rays. Its exact shape would depend on the long-term trend in incidence both of (single) X/ray treatments and of leukaemias among the spondylitics. If all non-leukaemic patients had been irradiated many times and if leukaemia was independent of irradiation, the percentage scores for the shortest intervals would be the highest, for

Table 6. *Sums of percentage scores as in Table* 4, *but for eleven "doubtful" cases only* (*cases* 1,2 3, 27, 29, 32, 37–41)

	Intervals (months)														
	0–11	12–23	24–35	36–47	48–59	60–71	72–83	84–95	96–107	108–119	120–131	132–143	144–155	156–167	Total
(a)	0	2	4	15	8	7	7	7	7	7	7	7	7	7	100
(b)	2.8	17.1	150.7	300.1	294.8	64.6	42.5	35.6	16.1	0	38.3	37.8	41.8	58.2	1100.4 (1100)
(β_1)	160	161	161	160	136	119	38	29	28	27	23	23	20	15	1100
(β_2)	20.8	71.5	145.2	336.2	258.3	91.5	33.0	28.2	26.6	25.0	19.6	18.0	14.3	10.6	1098.8 (1100)

Table 7. The same as Table 6, three cases of chronic leukaemia

(b)	0	28.4	0	100.0	100.0	25.6	46.0	—	—	—	—	—	—	300
(β_1)	45.6	43.5	43.5	42.5	21.4	21.4	21.4	17.7	5.2	5.2	5.2	5.2	5.2	
(β_2)	4.5	14.6	41.8	88.5	23.1	20.7	17.1	5.0	5.0	5.0	5.0	5.0	5.0	

reasons similar to those holding good for a uniform distribution of latent periods. The observed percentage scores for the eleven "doubtful cases", for the first few ranges of intervals, 0 to 11 months, 12 to 23 months, . . . up to 47 months, actually increase rapidly instead of decreasing, and look very similar to those for all thirty-one cases; this can be clearly seen by comparing observed and expected scores in Table 6. In fact the distribution of these eleven is even more sharply peaked. I have not been able to carry out a significance test because the distribution of these scores in random samples is so complex, but the evidence certainly favours most of the doubtful cases being causally related to irradiations. This kind of test should answer the doubts expressed by BRUES and by ABBATT and LEA[9] about applying dose–response analyses to these results.

The only obvious difference between the two classes relates to the longer time intervals. There are three cases with onset more than 10 years after the last dose and these are included among the "doubtful" ones. None of the three, however, was classed as doubtful by COURT-BROWN and DOLL.

The observed and expected scores for the three chronic myeloid leukaemias (Table 7) also show no obvious differences from the others. In the chronic cases the observed scores may be less reliable since the date of onset may be some months before the date of diagnosis.

7. THE DANISH SURVEY OF LEUKAEMIA CASES FOLLOWING IRRADIATION BY X-RAYS

A much larger series has been studied by FABER[6,14]. This consists not only of ankylosing spondylitics, but of all leukaemias in Denmark between 1942 and 1953. Cases with histories of therapeutic radiation and diagnostic radiation were separately compared with irradiated percentages in controls from the general population, with a similar age distribution. There was a greater percentage of acute and of chronic myeloid leukaemias that had been irradiated, but no higher percentage of chronic lymphatic leukaemias. When there was one course of irradiation more than 60 months before onset, FABER found little or no difference between the percentage of irradiated acute leukaemias and the corresponding percentage irradiated in controls; for chronic myeloid leukaemias there were relatively more cases than controls when the time interval between (one course of) irradiation and onset was less than 140 months. Hence the results suggest that the risk of developing acute leukaemia falls off after 5 years from irradiation, but that for chronic myeloid leukaemia it decreases more slowly.

Among the chronic myeloids in Denmark, about 30 had courses of treatment over a short period, or single courses. For these patients the most frequent time intervals were between 48 and 72 months (twelve cases). This is later than the maximum frequency for the eight chronic myeloids (including five more recent ones) so far established among the spondylitics in the British series.

Summarizing, the Danish results differ in at least three respects. There is certainly a pronounced maximum in the distribution of time intervals for the chronic myeloids, at about 60 months or higher. In chronic leukaemia the disease progresses slowly, so that the first irreversible changes in blood-forming organs in a chronic leukaemia may take place earlier relative to the manifestation of symptoms than they will in the acute leukaemias. The position of this maximum is also fairly near that for the maximum incidence of chronic myeloids at Hiroshima (see, for example BURCH[11] or COBB *et al.*[12])

On the other hand, for the acute leukaemias following therapeutic irradiation in Denmark

the most frequent interval between irradiation and onset was about 2 years. This is quite different from the spondylitics in Britain. The only obvious difference is that the Danish patients were being treated for different diseases and the sites of irradiation were therefore usually different. Also the average age was about 10 years higher; but in neither series does the latent period seem to be influenced by age, nor by the sizes of the doses. Thus there is no certain explanation for these differences.

8. THE MAXIMUM INCIDENCE OF CHILDHOOD LEUKAEMIA IN 3-YEAR-OLDS

Among children the incidence of leukaemia has more than doubled in the last 20 or 30 years, in several countries. The maximum incidence in Britain is among 3-year-olds.[10] A similar age distribution is found in America and in Holland. This maximum scarcely existed in the 1920's;[15] in Britain, for the decade 1931–1939 HEWITT[10] gives twenty-six per million for babies under 1 year, twenty-seven for 1-year-olds thirty-two for 2-year-olds, thirty for 3-year-olds and twenty-eight for 4-year-olds. The corresponding figures for the years 1950–1957 are 25.4, 34.2, 52.4, 64.6 and 49.5 respectively (all per million per year).

Data from the Oxford survey [7] show that for first-borns there is higher incidence, and an even more pronounced maximum among the 3-year-olds, than for the later children. (Full details given me by Dr. WEBB show that both differences are statistically significant*.)

The maximum of the distribution is at about 39 months; the corresponding maximum for cohorts is perhaps 1 or 2 months later. In any case this maximum is at about 4 years from the beginning of pregnancy. At these ages response to an insult may occur more quickly than in middle age: at least the processes are unlikely to be slower. Clearly it is possible that the maximum among the 3-year-olds is related to an insult during pregnancy. This would mean that such an insult, of whatever kind, has become much more frequent over the last 30 years.

9. LEUKAEMIA IN CHILDREN FOLLOWING IRRADIATION IN INFANCY OR IN UTERO

STEWART *et al.*[7] studied most of the cases of childhood leukaemia occurring in Britain in 1953, 1954 and 1955, amounting to about 700. About 10 per cent of these children had been irradiated by X-rays before birth or during infancy; they conclude that about half of these cases must be ascribed to this radiation. This of course still represents only a small increased risk to the child—a risk of about 1 in 1000—following abdominal X-rays given to the pregnant mother, but this risk is estimated to have been nearly doubled.

The intervals between irradiation and onset are given in Tables 8 and 9. Most of the cases and controls had been irradiated only once, or more than once but within a short period. Each case was matched by a control of the same sex and living in the same district, and with as nearly as possible the same date of birth. A few cases and controls with a history of several irradiations not close together in time, were left out. The intervals are those between irradiation and death for cases, and between irradiation of a control and death of the corresponding case. (Most childhood leukaemias are acute so that the intervals between onset and death are short.)

For children of 3 years and over when X-rayed for the first time, there seemed to be no difference between numbers of cases and controls X-rayed. The comparisons between irradiation under 3 years but not *in utero*, and *in utero* but not after birth, are given separately. The difference in numbers of cases and controls for the various intervals gives perhaps the best broad impression of the latent-period distribution. The intervals between irradiation and death seem to be shorter for irradiation after birth; the mode (most frequently occurring interval) is at about 24 months. For X-rays *in utero* the mode is greater than 60 months. There are, of course, large random variations.

The column giving the "expected" numbers

* It is not clear whether this is being allowed for in analysing the corresponding survey in California. In the data quoted by KAPLAN[16], comparing abdominal X-rays of pregnant mothers among cases and controls, both siblings and playmates were used as controls. If children with no brothers and sisters were not included this would bias the results.

Table 8. *Intervals between irradiation and death for children in the Oxford survey irradiated before the third birthday, but not* in utero

Months	Cases	Controls
0–11	8	6
12–23	10	2
24–35	7	1
36–47	5	4
48–59	6	3
60–71	1	0
72–83	3	0
84–95	1	0

of controls is of interest. This is what the frequency distribution of thirty-seven intervals between 1 month before birth and death would be (1 month because most pre-natal irradiations took place round about 1 month before birth), if the distribution of ages at death of the cases paired to these controls is the same as that for all 700 in the survey. The observed numbers of controls for the longer intervals are all less than the "expected" ones on this basis. HEWITT points out the main reason for this. All deaths were between 1953 and 1955, so that the longer intervals correspond to children born earlier. The overall incidence of X-rays to infants and pregnant mothers increased considerably between about 1944 and 1950.

Table 9. *Similar analysis to Table* 8; *irradiation* in utero *only; observed irradiation among controls and expected numbers if the distribution of intervals corresponds to the ages at death of the whole population of child leukaemias*
(See text)

Months	Cases	Controls observed	Controls "expected"
0–11	4	3	1.8
12–23	7	6	3.2
24–35	10	8	5.6
36–47	10	6	6.2
48–59	7	8	5.0
60–71	8	2	4.1
72–83	3	2	3.7
84–95	5	0	2.8
96–107	2	0	2.4
108–119	3	1	1.9
120–121	0	1	0.3
Total	59	37	37

The general conclusion, applying the concept of latent period to the incidence in children, is then that if the maximum among 3-year-olds is due to an "insult" at conception or early in pregnancy, the distribution of time intervals between insult and onset are similar for children, although less sharply peaked at 4 years. How closely the distributions correspond will not be known until the final analysis is done. For the children actually irradiated (*in utero* or after birth) the correspondence is only moderately close. On balance the evidence indirectly supports the hypothesis that an insult to the foetus is more dangerous than one received later in life, and irradiation by X-rays is a possible insult but not the only one.

10. THE RELATION BETWEEN DOSE OF IONIZING RADIATION AND INCIDENCE OF LEUKAEMIA

The original idea of studying the time intervals between irradiation and onset of leukaemia was simply to obtain more information from the data. Time, being a continuous variable, ought to provide a better measure of response than one limited to "all or nothing", presence or absence of leukaemia. The dose–response relationships are needed as much as before, and it is planned to recalculate them allowing for as good an estimate of the distribution of latent periods as we can get, based on information for four more years. For this calculation, dose records are needed also for a fair-sized sample of "non-leukaemias" in order to estimate the number of man-years at risk.

The dose–response curves obtained by assuming a uniform distribution of latent periods may be quite wrong. For example, in about two-thirds of the series of leukaemias in ankylosing spondylitics, the disease was diagnosed within 3 years from the last treatment. From our analysis, the probability of onset within 3 years is several *times* less than its values in the third

and fourth years. Then, in calculating a dose–response curve, the dose given 2 years before onset must be multiplied by a factor small compared with the one for a dose 4 years before onset. Not allowing for this introduces, in effect, an error of several hundred per cent into very many of the laboriously obtained estimates of doses, whether dose to the spinal marrow is chosen, or the whole body integral dose, or any other kind.[6]

Clearly, then, we do not yet know how incidence is related to dose, or even the shape of the dose–response curve. COURT-BROWN and DOLL[17] had warned that their suggested linear relationship was only provisional, but several recent studies are based on it, and the conclusions in them might therefore have to be changed.[18–20] On the other hand BRUES[13] and BURCH[11] maintain that a non-linear dose–response curve fits equally well or better; a similar warning against drawing firm conclusions holds equally for these studies (as BURCH pointed out in relation to his own work).

Some ideas on the mechanism of leukaemia induction might have to be modified. A linear variation of incidence with dose is compatible with a "random-hit" model, whereby each quantum of radiation has an equal, but very small, probability of producing a lasting change in a living cell (such as a "somatic mutation"); this probability per quantum could vary among individuals.[21,22] A maximum risk to the individual during the fourth year suggests that the damaged parts begin to recover in the fifth year; possibly leukaemia develops when healing processes get out of control, rather than as a direct consequence of radiation damage. HEWITT[10] suggests that a secondary stimulus may be at work, and he points out that there are large non-random and non-seasonal fluctuations in the incidence in the whole population from year to year, both in adults and in children,[15] which could mean that many people have been exposed to an external stimulus at about the same time.

COURT-BROWN and DOLL have worked out a dose–incidence table for a very uneven distribution of latent periods, with relative probabilities 0 in the first year, 0.1 in the second, 0.5 in the third, 1.0 in the fourth and fifth, 0.5 in

Table 10. *Reproduced from* COURT-BROWN *and* DOLL[3] *Table* 4D
(Data on the incidence of leukaemia among the sample patients, for the distribution of latent periods given in Section 10)

		Mean equivalent dose to spinal marrow (r)						
	0*	Less than 100	100–249	250–499	500–749	750–999	1000–1249	1250 or more†
No. of man-years exposed to risk	—	20,452	12,568	11,033	7563	4623	3649	2177
No. of cases of leukaemia	—	5	3	6	11	4	4	4
Incidence per 10,000 men per year	0.49	2.44	2.37	5.44	14.54	8.65	10.96	18.37

* The rate given for "zero" dose represents the corresponding incidence among men of the same age-distribution as those in the study series and observed over the same period, calculated from the mortality from leukaemia experienced by the whole male population of Britain.

† Mean dose, 1445 r.

the sixth and 0.1 in all later years. This is reproduced here (Table 10). This distribution was suggested by inspecting the intervals between single doses and onset (as can be seen from Table 1). It is much nearer the estimates given in this paper than it is to a uniform distribution. Probably, therefore, Table 10 provides the best available dose–response relationship at present. There is still an evident increase of risk with dose, but it is very irregular. Feeding in more results and the best estimate of latent periods, when this is known, may of course, lead to a smoother curve.

Another possibility is that the measures of dose chosen are the wrong ones for assessing risks to the human body. VENNART[23] suggests that other parts of the body, or of the marrow, may be more sensitive to radiation than the spinal marrow. We have studied the full case records (where available) of the leukaemias; in nearly all of them the spleen, for example, could have been irradiated, through irradiation of the whole spinal chord, or the thorax. In the few instances of irradiation to the sacro-iliac joints only, there were other doses which, allowing for the latent-period concept, were more likely to have induced leukaemia. This idea can be tested only by a fuller analysis of case records for irradiated non-leukaemias. If it is true that some methods of irradiation are more dangerous than others, treatment by X-rays might be made safer without losing its effectiveness.

Among other variables that might affect the results is the dose rate. MOLE[24] finds this important when irradiating mice. LAMERTON[25] points out the importance of protraction and fractionation.

If one or all of these new variables affect the risks, apart from their correlation with the total dose, that fact does not in itself invalidate any analysis that ignores them, unless they are systematically related to the dates and intensities of the doses. If not, their effect will be to increase an apparently random scatter in observed dose–response curves.

In any event it seems more firmly established than before that most of the cases arise from irradiation by X-rays, since if many cases occurred independently, the distribution of latent periods would be more nearly uniform. Similarly, a mixture of cases caused and not caused by radiation will have a distribution of latent periods that is a superposition of this nearly uniform one and one with a fourth-year maximum. This kind of mixed distribution should be present in FABER's very valuable series, which is being studied further.

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FALLOUT AND NATURAL BACKGROUND IN THE SAN FRANCISCO BAY AREA

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Abstract—On or about 20 March 1958 a significant increase in the background radiation levels was noticed at the University of California Radiation Laboratory. It was soon determined that this increase was associated with radioactive material deposited during a series of rainstorms. The Health Physics department began a series of measurements in the San Francisco Bay area in order to contribute to the assessment of the extent and amount of the deposited radioactivity.

The instrument used for the field surveys was a NaI(Tl) crystal 3 in. long by 3 in. in diameter and viewed by a DuMont K1197 phototube, the output of which drives a count-rate meter.

This same crystal was used for γ-ray pulse-height analysis in conjunction with a 50-channel pulse-height analyzer. Sample counting is done inside of a 4 in. thick lead shield.

In making the survey, the points were chosen to be as evenly spaced and as representative of altitude, rainfall and rural and urban areas as possible.

One area, an area of dense fern growth, showed the highest radiation levels and it was decided to use this fern for pulse-height analysis. Samples were ashed and then analyzed. The results of this analysis did bear out the belief that this was fallout-produced radiation. The prominent peaks at 0.15, 0.49 and 0.75 MeV are characteristic of fallout spectra of this age. The 1.62 MeV La^{140} peak was also present.

The results of the several surveys enabled us to establish a value for the natural background radiation for these areas as none was available prior to the deposition of the fallout.

By the use of γ-ray spectroscopy in conjunction with a γ-ray dose field survey, it is possible to measure changes in the radiation level due to fallout that in magnitude do not exceed 10 per cent of the natural background. A continuing program is being carried out both in the field surveys and the spectroscopy.

I. INTRODUCTION

On or about 20 March 1958, a significant increase in the background radiation level was noticed at the University of California Radiation Laboratory. It was soon determined that this increase was associated with radioactive material deposited during a series of rainstorms. In order to be able to contribute to the assessment of the extent and amount of deposited radioactivity, the Health Physics Department began a series of γ-radiation measurements in the San Francisco Bay area.

II. INSTRUMENTATION

The instrument used for the field surveys was a NaI(Tl) crystal 3 in. long by 3 in. diameter viewed by a DuMont K1197 phototube, the output of which feeds a count-rate meter. The count-rate circuit is biased to exclude all pulses corresponding to a γ-ray energy below 0.03 MeV. The brass tube enclosing the crystal and phototube has walls 1.36 g/cm^2, thus effectively excluding β-particles below 2.75 MeV.

The reason for deciding to exclude β-particles was twofold. First, a thin-walled plastic ion chamber was used simultaneously with the NaI counter at a few locations, and there was not a constant ratio between the readings. The plastic ion chamber gave from two to five times the reading of the NaI counter. We interpreted this to mean that the portion of the

β-dose seen by the plastic ion chamber was strongly dependent on the immediate environment. Second, for small doses the gonadal region of the body is most important, and external β-radiation is not effective in contributing to gonadal dose.

For γ-ray pulse-height analysis the same crystal mentioned above was used in conjunction with a 50-channel pulse-height analyzer of UCRL design. In the calibration of the pulse-height analyzer, channel-width equality is checked by a special fixed-frequency pulse generator whose output pulse height raises linearly with respect to time; therefore channels of equal widths receive equal numbers of counts. Both linearity and resolution of the entire detection system are carefully measured with a Na^{22} source which emits two γ-rays (0.510 MeV and 1.277 MeV) in an energy range convenient for this work. Fig. 1 shows a typical source calibration curve.

All sample counting is done inside a 4-in.-thick lead shield so that the detector response to ambient background radiation is greatly suppressed. The lower curve of Fig. 1 shows a typical profile of the residual background (BKG) in this energy region.

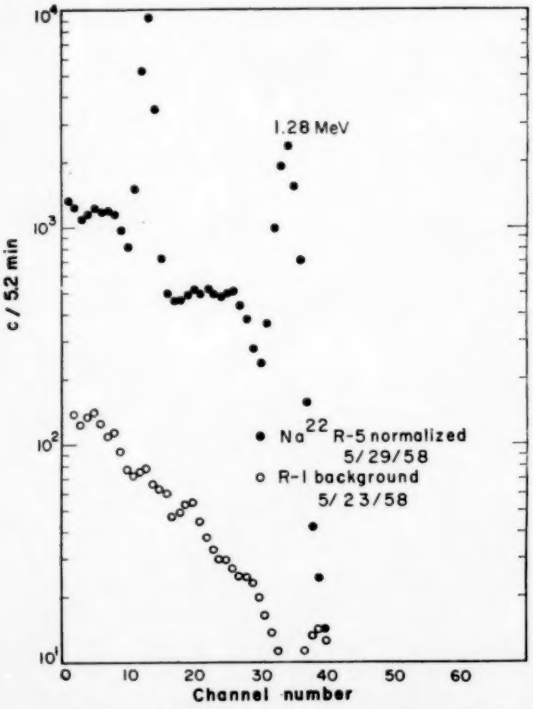


FIG. 1. Pulse-height analysis showing peaks from Na^{22} γ-rays and crystal background.

III. METHOD

The individual survey points were chosen to be evenly spaced and representative of the various conditions in regard to altitude, rainfall and plant growth found in urban and rural areas near San Francisco Bay. It is believed that, except for the fallout, no man-made radiation sources influenced the readings taken. At each location the detector was positioned 3 ft above the ground, and an attempt was made to determine the average of the minor variations in radiation intensity. Fig. 2 is a map showing the location of the individual survey points. The samples of dried and fresh fern were taken at Location 24. Sample no. 1 was taken on 29 March, ashed and then analyzed. It was dried fern, presumably having grown, matured and died before the fallout event. Sample no. 2 was taken on 17 June, and analyzed before and after ashing. It was also dried fern from the same crop. Sample no. 3 was taken on 9 July and ashed before analysis. It is this year's crop, having begun growth in April and now mature.

No data were available on the natural-background radiation level for any of the locations before the fallout occurred. Therefore, it is necessary to establish a background level for each location so that the added increment due to the fallout can be estimated. First we must set a day on which the fission products were formed. (Data taken with the pulse-height analyzer confirmed the supposition that the fallout radioactivity was due to fission products.)

Table 1 gives information regarding Russian nuclear weapons tests as announced by the U.S. Atomic Energy Commission.

Since the fallout here at the Laboratory began on 20 March, it will be assumed that the fission products

Table 1

Date	Location	Size
14 March, 1958	North of Artic circle	Small
14 March, 1958	Siberia	Small
15 March, 1958	Siberia	Small
20 March, 1958	Siberia	Small
21 March, 1958	North of Artic circle	Larger
22 March, 1958	Siberia	Medium

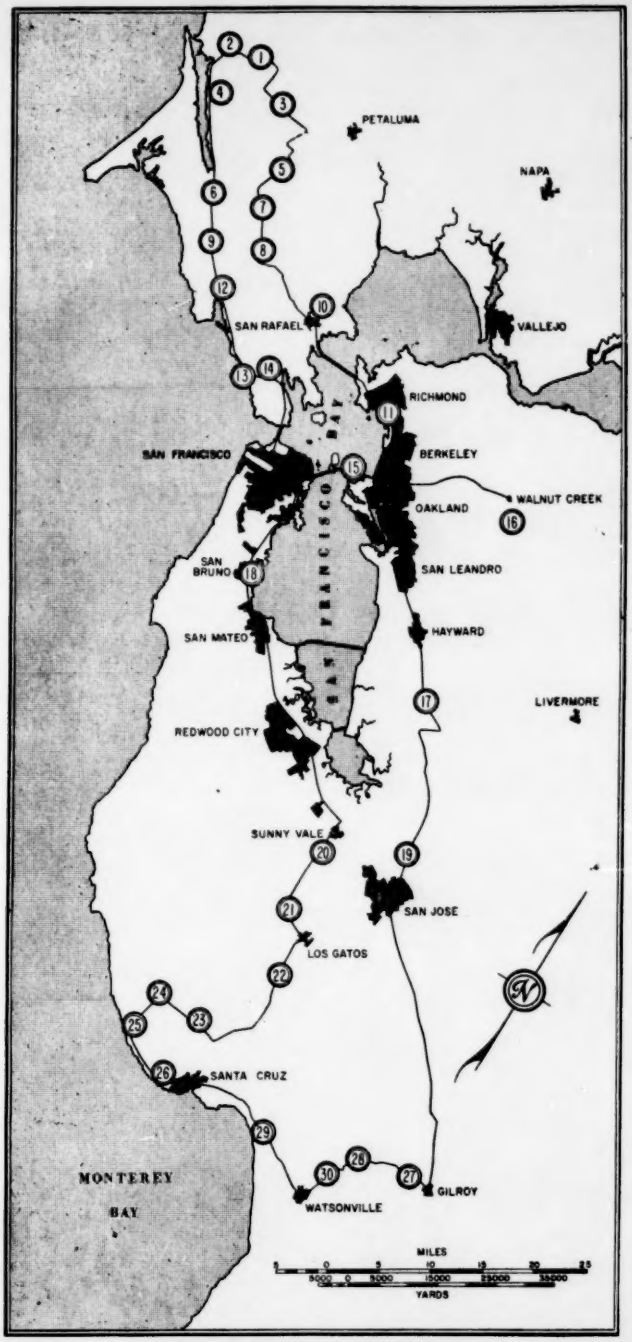


FIG. 2. Map of survey area.

were formed on 15 March. Next, it is necessary to find a period during which normal fission-product decay occurs.

Table 2 gives data regarding the dates of the field surveys and the calculated percentage of fission products remaining, based on the assumption that 100 per cent were formed on 15 March and the usual decay formula $R_t/R_0 = t^{-1.2}$, operates

Table 2

Date of Survey	Age of fission products (days)	Percentage remaining (%)
28 March, 1958	13	4.6
29 March, 1958	14	4.2
10 April, 1958	26	2.0
11 April, 1958	27	1.9
8 May, 1958	54	0.82
9 May, 1958	55	0.82
16 June, 1958	93	0.44
17 June, 1958	94	0.44

Fig. 3 gives the amount of rainfall received here at UCRL from 20 March through 30 June. No rain fell from 10 April through 9 May and information given us by MYRON THAXTER, Health Chemistry Department, and MARGARET WHITE, Donner Laboratory, indicates that little if any additional fallout was deposited during this period. Consequently, perturbation of normal fission-product decay should be minimal.

The background intensity of γ-radiation for each survey point was then determined in the following manner. Let A equal total radiation intensity at a location; b, background intensity; and f, fallout intensity. Assume f will decay by a factor of 2.44 (2.0/0.82) between 10 April and 8 May and a factor of 2.32 (1.9/0.82) between 11 April and 9 May. Then we have

$$A_{10\,\text{April}} = b + 2.44f$$

$$A_{8\,\text{May}} = b + f$$

$$f = \frac{(A_{10\,\text{April}} - A_{8\,\text{May}})}{1.44}$$

and

$$b = A_{8\,\text{May}} - \frac{(A_{10\,\text{April}} - A_{8\,\text{May}})}{1.44}$$

Similarly for the other pair of survey dates we have

$$b = A_{9\,\text{May}} - \frac{(A_{11\,\text{April}} - A_{9\,\text{May}})}{1.32}$$

IV. RESULTS

Table 3 gives the results of the field surveys. At the following locations it was necessary to use an average background of 0.0070 mr/hr because of known perturbation in normal fission-product decay at these points. At Location no. 16 measurements were made on 4, 5 and 6 April during a series of rainstorms; Locations no. 18 and no. 22 are drainage areas into which fallout was carried by surface water, and Location no. 28 was inaccessible on 11 April survey because of a landslide.

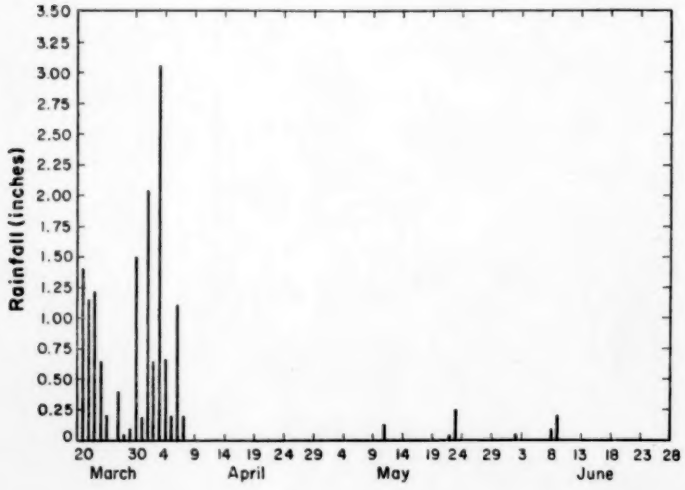


FIG. 3. Graph showing inches of rainfall per day in 1958 at UCRL.

Table 3. Fallout increment above background

Location	Background mr/hr	23–29 March mr/hr	23–29 March c/mile²	10–11 April mr/hr	10–11 April c/mile²	8–9 May mr/hr	8–9 May c/mile²	16–17 June mr/hr	16–17 June c/mile²
1	0.0052	0.0168	4.00	0.0190	4.52	0.0078	1.86	0.0077	1.83
2	0.0083	0.0237	5.64	0.0225	5.36	0.0092	2.19	0.0073	1.74
3	0.0071	0.0229	5.45	0.0181	4.32	0.0074	1.76	0.0058	1.38
4	0.0099	0.0161	3.83	0.0149	3.55	0.0061	1.45	0.0032	0.76
5	0.0091	0.0149	3.55	0.0139	3.31	0.0057	1.36	0.0048	1.14
6	0.0068	0.0242	5.76	0.0212	5.05	0.0087	2.07	0.0069	1.64
7	0.0064	0.0256	6.09	0.0258	6.14	0.0106	2.52	0.0079	1.88
8	0.0066	0.0244	5.81	0.0254	6.05	0.0104	2.48	0.0086	2.05
9	0.0088	0.0222	5.28	0.0200	4.76	0.0082	1.95	0.0060	1.43
10	0.0045	0.0275	6.55	0.0255	6.07	0.0105	2.50	0.0091	2.17
11	0.0046	0.0234	5.57	0.0242	5.76	0.0099	2.36	0.0083	1.98
12	0.0124	0.0146	3.47	0.0112	2.67	0.0046	1.09	0.0028	0.67
13	0.0099	0.0161	3.60	0.0125	2.98	0.0051	1.21	0.0035	0.83
14	0.0068	0.0142	3.38	0.0200	4.76	0.0082	1.95	0.0063	1.50
15	0.0066 0.0074 0.0064	0.0124 0.0106 0.0116	2.95 2.52 2.76	0.0120 0.0107 0.0120	2.86 2.55 2.86	0.0052 0.0044 0.0052	1.24 1.05 1.24	0.0024 0.0028 0.0026	0.57 0.67 0.62
16	0.0070	0.0210 0.0160	5.00 3.81	0.0180	4.29	0.0100	2.38	0.0075	1.78
17	0.0065	0.0135	3.21	0.0155	3.69	0.0067	1.59	0.0066	1.57
18	0.0070	0.0140	3.33	0.0710	16.9	0.0208	4.95	0.0078	1.86
19	0.0075	0.0110	2.62	0.0109	2.59	0.0047	1.12	0.0039	0.93
20	0.0073	0.0127	3.02	0.0123	2.93	0.0053	1.26	0.0041	0.98
21	0.0054	0.0226	5.38	0.0200	4.76	0.0086	2.05	0.0077 0.0074	1.83
22	0.0070	0.0140	3.33	0.0350	8.33	0.0090	2.14	0.0071	1.69
23	0.0066	0.0254	6.05	0.0302	7.19	0.0130	3.09	0.0105	2.50
24	0.0097	0.0323	7.68	0.0355	8.45	0.0153	3.64	0.0121	2.88
25	0.0076	0.0134	3.19	0.0160	3.81	0.0069	1.64	0.0055	1.31
26	0.0062	0.0188	4.47	0.0186	4.43	0.0080	1.90	0.0069	1.64
27	0.0063	0.0147	3.50	0.0179	4.26	0.0077	1.83	0.0075	1.78
28	0.0070	0.0225	5.36			0.0085	2.02	0.0091	2.17
29	0.0076	0.0154	3.66	0.0160	3.81	0.0069	1.64	0.0048	1.14
30	0.0093	0.0157	3.74	0.0121	2.88	0.0052	1.24	0.0045	1.07
Average		0.0183		0.0205		0.0082		0.0063	

The values of c/mile² were determined from data given by GLASSTONE[1] (pp. 435–436) namely, 3 ft above the ground whose surface is evenly contaminated to the extent of 1 Mc/mile² with a γ-emitter of 0.7 MeV energy, the dose rate is 4.2 r/hr.

The results of the γ-ray pulse-height analysis of the samples from Location 24 are shown in Figs. 4–7.

The upper curve, Fig. 4, shows the spectrum from Sample no. 1, 0.91 g of brown-fern ash, collected on 29 March 1958 and counted on 13 May 1958. The prominent peaks at 0.15, 0.49 and 0.75 MeV are characteristic of fallout spectra of this age. The 1.62 MeV La^{140} peak is also prominent but cannot be shown on this scale plot.

The lower curve of Fig. 4 shows the residual BKG in this energy range. Use these two curves, and make the assumption that an activity can be determined in a reasonable counting time when it produces a count rate

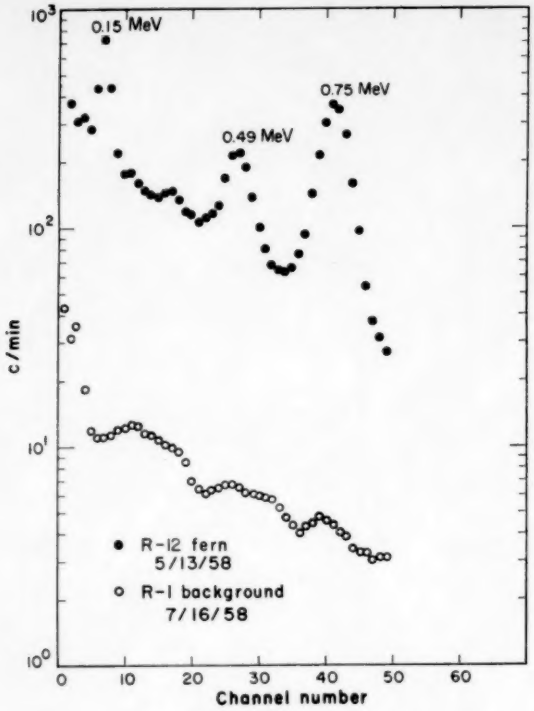


FIG. 4. Pulse-height analysis of Fern Sample no. 1 on 13 May 1958 showing fission-product peaks and crystal background.

equal to the BKG count rate. Then the minimum concentration of fallout detectable from a sample similar to this (0.9 g of brown-fern ash) can be shown to be 0.04 c/mile^2 when only the 0.75 MeV peak is observed, and can be shown to be 0.11 c/mile^2 when all three prominent peaks are observed—essentially positive identification.

In Fig. 5 the upper curve shows the γ-spectrum obtained from the same brown-fern ash counted on 25 July 1958. The lower curve again shows residual BKG. Fig. 6 shows the spectrum obtained from a 0.44 g sample of brown-fern ash (Sample no. 2), collected on 17 June 1958 and counted on 25 July 1958; a BKG curve also appears on this figure. Fig. 7 shows the spectrum obtained from a 1.41 g sample of green-fern ash (Sample no. 3), collected on 9 July 1958 and counted on 25 July 1958; again a BKG curve is shown on this figure. The green-ash sample count is seen to be only slightly above BKG level; however, all three peaks are clearly defined. The fallout spectra can be further clarified by subtracting the BKG curve from the fern-sample curve. This technique has not been used here because statistical errors in fern-sample data points do not permit generation of a smooth difference curve. A longer counting time would be necessary to obtain results of quantitative value.

In Fig. 8 are plotted the curves for the $t^{-1.2}$ law and the thirty-location average of the fallout levels (total reading − calculated background). Notice that the center portions of curves A and B fit the assumed decay curve because they were forced to do so by the method used in calculating the background for each location. Curve A is drawn on the assumption that the fallout was deposited smoothly in time, and the area under it

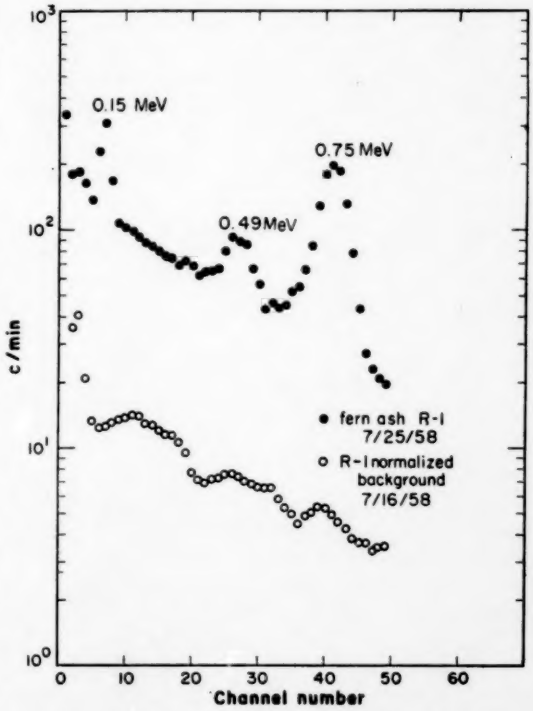


FIG. 5. Pulse-height analysis of Fern Sample no. 1 on 25 July 1958 showing fission-product peaks and crystal background.

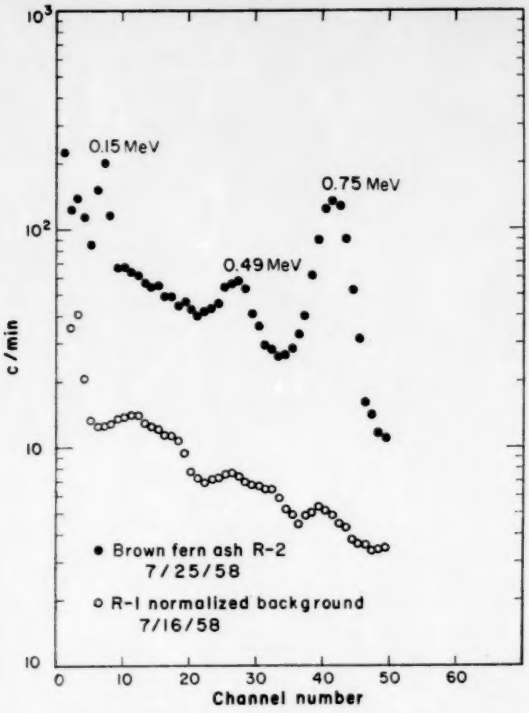


FIG. 6. Pulse-height analysis of Fern Sample no. 2 on 25 July 1958 showing fission-product peaks and crystal background.

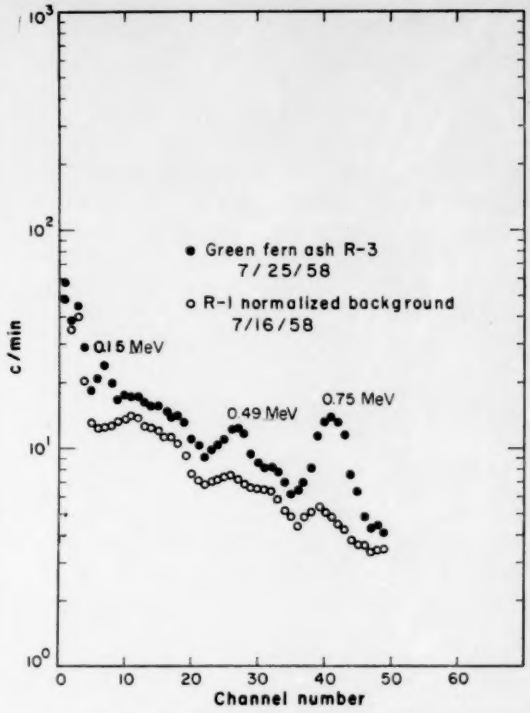
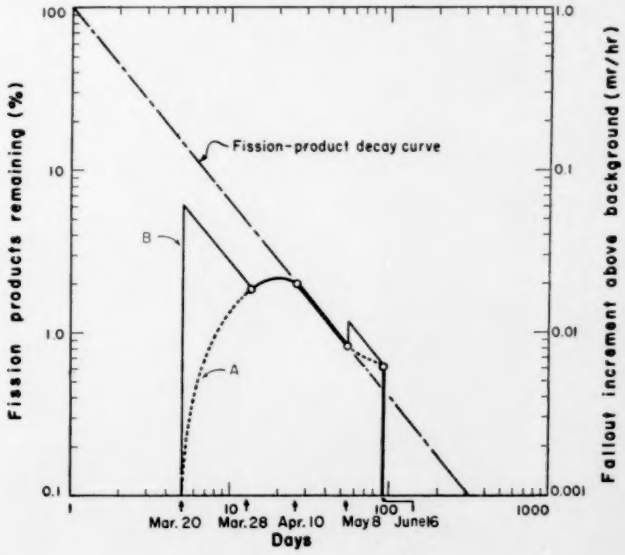


FIG. 7. Pulse-height analysis of Fern Sample no. 3 on 25 July 1958 showing fission-product peaks and crystal background.

FIG. 8. Curves showing fission-product decay and the average fallout increment above background for all thirty locations.



probably represents a lower limit of integrated exposure. Curve B is drawn assuming a sudden deposition of fallout because of rain on 20 March and 11 May. It is drawn smoothly between 28 March and 10 April because, although additional fallout was deposited during this period, the rainfall was extremely heavy and immediate run-off of rain water containing fission products was probably important. Curve B represents a probable upper limit of integrated exposure. For a 24 hr/day outside exposure, the γ-dose between 20 March and 17 June is given by

$$d = \int_{20\ \text{March}}^{16\ \text{June}} n(A)\ dt$$

A graphical integration under curve A gives 24 mr and under curve B 28 mr, and in each case this is about one-half the dose delivered in an infinite time subject to actual radiological decay and local weathering effects.

It is of interest to compare the ratio of actual fern activities with the ratio of the actual and predicted measurements made at all locations and the ratio of the actual and predicted measurements made where the ferns were collected. On 25 July, Sample no. 2 (collected 17 June) gave 295 c/min per g of ash and Sample no. 1 (collected 19 March) gave 209 c/min per g of ash, or a ratio of 1.41. The all-location average ratio (from Fig. 8) is

$$\frac{0.0063}{0.0044}\left(\frac{\text{actual}}{\text{predicted}}\right)$$

or 1.43, and the ratio for Location 24 alone is

$$\frac{0.0121}{0.0085}\left(\frac{\text{actual}}{\text{predicted}}\right)$$

or 1.42. The agreement, although better than expected, tends to give us confidence in our method of background calculation and in our surveying technique. As can be seen from Fig. 8, normal fission-product decay would cause these ratios to be 1.0. We interpret the additional activity to be due to fallout from U.S. tests deposited during the rains of May and June.

CONCLUSION

By the use of γ-ray spectroscopy in conjunction with a γ-ray dose field survey, it is possible to measure changes in the radiation level due to fallout that in magnitude do not exceed 10 per cent of the natural background. It is planned to continue both the field surveys and the spectroscopy.

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CALCULATIONS OF MAXIMUM PERMISSIBLE CONCENTRATIONS OF RADIOACTIVE FALLOUT IN WATER AND AIR BASED UPON MILITARY EXPOSURE CRITERIA*†

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Abstract—Computations are tabulated for maximum permissible concentrations (MPC) of radioactive fallout in water and air for consumption during selected periods at designated times after burst to meet specified dose criteria, namely, 15 rems in 90 days and 150 rems in 30 days. The MPC values obtained may be used directly to obtain values for other doses during the same periods of 30 and 90 days by simply using the appropriate factor. For example, the MPC for a dose of 1.5 rem in 90 days is one-tenth the tabulated value for 15 rems.

The fourteen major radionuclides of the fallout are evaluated in terms of their individual contributions under varying situations to the radioactivity of the fallout mixture. Only limited, unofficial MPC data have been available for single exposures, hence in the computations of this paper, emphasis has been given to exposure periods ranging to 90 days. The calculations for radioactive fallout are based upon the critical organ yielding the smallest MPC for each element.

The study shows that, in general, the principal contributors in the fallout mixture to the dose to internal body parts during the first seven days after burst are: iodine-131, neptunium-239, barium-140 + lanthanum-140, strontium-89, zirconium-95 + niobium-95, and yttrium-91. The main contributors from 8 to 105 days after burst are; iodine-131, strontium-89, cerium-144 + praseodymium-144, barium-140 + lanthanum-140, and yttrium-91, whereas strontium-89, cerium-144 + praseodymium-144, zirconium-95 + niobium-95, strontium-90 + yttrium-90 and yttrium-91 are the principal contributors from 105 days to 365 days after burst.

INTRODUCTION

MAXIMUM permissible concentrations (MPC's) of individual radioisotopes in water and air for continuous lifetime exposures or for the equilibrium condition (rate of elimination = rate of body deposition) are reported in the National Bureau of Standards Handbook 52 issued in 1953.[2] The basis for the MPC values in this internal dose handbook for chronic exposures was complemented in 1954 by the MPC values for single exposures to individual isotopes given by MORGAN and FORD[3]. In 1956, DICK *et al.*[4] proposed realistic MPC levels for mixed fission products at different times after burst and for several exposure periods. Their values were based upon a dose criterion of 0.3 rem/week. MPC values were not calculated for the total radioactive fallout or for dose criteria of interest to the military services.

The present study has not only considered MPC values based on military dose criteria but has also determined MPC's for the total radioactive fallout mixture. Calculations have been made for a number of exposure periods at various times after burst. Although the values reported here have been determined for certain specified military dose criteria, the method of

* A condensed revision of Technical Report-182 issued by the U.S. Naval Radiological Defense Laboratory.[1]

† The opinions or assertions contained in this article are the private ones of the authors and are not to be considered as official or reflecting the views of the Navy Department or the Naval Service at large.

calculation used can be applied to other dose criteria. The MPC values at the start of intake were calculated in terms of days of intake (exposure) period and eleven different times after burst. The military dose criteria for which MPC's of individual isotopes in water and air were calculated are: (1) 15 rems* in 90 days for intake periods of 1, 7, 14, 30, 60 and 90 days; and (2) 150 rems in 30 days for intake periods of 1, 7, 14 and 30 days. These MPC values were computed on the basis of exposures to the lower portion of the large intestine and other critical organs (bone, thyroid, muscle or kidney, depending on the radioisotope considered).

The MPC calculations are based on certain assumptions regarding such aspects of the problem as deposition of radioisotopes in tissue, solubility of radioactive compounds in mixed fallout, rate of biological elimination of radioisotopes, relative sensitivity of critical organs to radiation, and relation of partial-body effects to whole-body effects. Therefore, until more definitive data are available to enable modification of the MPC's given in this report, the values should be considered as good approximations only.

* A rem is equal to the product: RBE × rad, where RBE is the relative biological effectiveness, and the rad corresponds to an absorbed dose of 100 ergs/g of tissue.

METHODS OF ANALYSIS

The MPC values were calculated from the following equations:

$$(\mathrm{MPC}_w)_n = \frac{6.15 \times 10^{-6}\, m\Sigma D}{f_w N \Sigma(\mathrm{RBE}) ET \left\{ \dfrac{1 - \exp(-0.693\, n/T_r)}{1 - \exp(-0.693/T_r)} - \exp(-0.693 t/T) \left[\dfrac{\dfrac{1 - \exp[-0.693(T - T_r)n]}{TT_r}}{\dfrac{1 - \exp[-0.693(T - T_r)]}{TT_r}} \right] \right\}}$$

$$(\mathrm{MPC}_a)_n = \frac{6.77 \times 10^{-10}\, m\Sigma d}{f_a N \Sigma(\mathrm{RBE}) ET \left\{ \dfrac{(1 - \exp(-0.693\, n/T_r)}{1 - \exp(-0.693/T_r)} - \exp(-0.693 t/T) \left[\dfrac{\dfrac{1 - \exp[-0.693(T - T_r)n]}{TT_r}}{\dfrac{1 - \exp[-0.693(T - T_r)]}{TT_r}} \right] \right\}}$$

$$(\mathrm{MPC}_w{}^{\mathrm{g.i.}})_n = \frac{1.78 \times 10^{-5}\, m' G \Sigma D (1 - e^{-\lambda})}{\Sigma(\mathrm{RBE}) E (h_1 - h_0) f_w' (1 - e^{-\lambda n})}$$

$$(\mathrm{MPC}_a{}^{\mathrm{g.i.}})_n = \frac{2 \times 10^{-9}\, m' G \Sigma D (1 - e^{-\lambda})}{\Sigma(\mathrm{RBE}) E (h_1 - h_0) f_a' (1 - e^{-\lambda n})}$$

$(\mathrm{MPC}_w)_n$ = maximum permissible concentration in water and defined as the concentration of radioactivity in microcuries per milliliter in water at the start of the intake for intake period (n) that will result in a total absorbed dose (ΣD) over a period of (t) days to critical tissues other than g.i. tract.

$(\mathrm{MPC}_a)_n$ = maximum permissible concentration in air based on dose (ΣD) to critical tissues other than g.i. tract.

$(\mathrm{MPC}_w{}^{\mathrm{g.i.}})_n$ = maximum permissible concentration in water based on dose (ΣD) to g.i. tract.

$(\mathrm{MPC}_a{}^{\mathrm{g.i.}})_n$ = maximum permissible concentration in air based on dose (ΣD) to g.i. tract.

n = intake period (days).

m = mass of critical organ (g).

m' = mass of material in critical portion of g.i. tract = 150 g when $T > \frac{1}{2}$ day.

ΣD = total absorbed dose (rems).

f_w = fraction of ingested activity deposited in critical organ.

f_a = fraction of inhaled activity deposited in critical organ.

f_w' = fraction of ingested activity reaching the critical portion of the g.i. tract = $1 - f_1$ (f_1 is the fraction going from g.i. tract to blood).

f_a' = fraction of inhaled activity reaching the critical portion of the g.i. tract = $0.62(1 - f_1)$.

N = distribution factor.

(RBE) = relative biological effectiveness of the radiation.

E = effective energy of radiation of both the radioactive isotope and its daughters (MeV).

T = effective half-life (days).

T_r = radioactive half-life.

t = period from beginning of ingestion or inhalation to any designated time later in days ($t > n$).

λ = dissociation constant $= \dfrac{0.693}{T_r}$.

h_0 = time of arrival of radioactive isotope at critical portion of g.i. tract = 13/24 day when $T_r > \frac{1}{2}$ day.

h_1 = time of departure of radioactive isotope from critical portion of g.i. tract = 31/24 day when $T_r > \frac{1}{2}$ day.

$$G = \frac{0.693(h_1 - h_0)}{[\exp(-0.693h_0/T_r) - \exp(-0.693h_1/T_r)]T_r} = 1, \quad \text{where } T_r \gg 1.$$

As shown in the Appendix these equations are modifications of those used by MORGAN and FORD[3] for expressing: (1) the dose, for a single exposure (1 day) to contaminated water or air, to critical organs other than the g.i. (gastrointestinal) tract [equation (A.1) of the Appendix], and (2) the dose, for a single exposure (1 day) to contaminated water or air, to the g.i. tract.

Calculations were first made of the MPC's of individual isotopes in water and air required to produce the criteria doses (Tables 3 and 4). The MPC value obtained for each isotope was then divided by the fractional activity of the isotope (that is, the ratio of isotope activity to total activity) at a number of times after burst. A typical example of values obtained is shown in Table 5 for $Sr^{90} + Y^{90}$. Finally, the harmonic mean (reciprocal of the sum of the reciprocals) of the values of the isotopes considered was taken to obtain the MPC's of the mixed radioactive fallout in water and in air (Tables 6 and 7).

The fractional activities of the fallout radioisotopes were determined from HUNTER and BALLOU[5] for the thirteen most hazardous fission products. The fractional activity of Np^{239} in the mixture was calculated from the peak value at 4 days (60 per cent) reported by STEWART *et al.*[6] (Table 2).

The values of all the factors used in the equations except those for N are listed in Table 1. The value of the parameter, N, was taken to be 5 for all bone-seeking isotopes and unity for all others, as recommended by the International Commission on Radiological Protection.[7]

ASSUMPTIONS

Deposition

Since dose to tissue is expressed as absorbed energy per unit weight of tissue, calculation of the dose cannot be made without knowledge of the weight of tissue receiving the radiation from the deposited radioisotope. The weight of that portion of the tissue that contains the deposited radioisotope is used in the dose calculations. The dose determinations made in this study are based on the assumption that the radionuclides are deposited homogeneously

Table 1. *Constants for calculating maximum permissible concentrations of radioisotopes in water and air*

Radioisotope	Effective energy, ΣE (MeV)	Organ of deposition		Half-life physical T_r (days)	Half-life biological T_b (days)	Effective T (days)	Fraction from g.i. tract to blood f_1	Fraction in C.O. of that in total body f_2	Fraction from blood to critical organ f_3	Fraction reaching critical organ	
		Organ	Mass (g)							Ingestion f_w	Inhalation f_a
Sr^{89}	0.55	Bone	7×10^3	53	3.9×10^3	52	0.6	0.7	0.4	0.25	0.22
$Sr^{90} + Y^{90}$	1.0	Bone	7×10^3	9.1×10^3	3.9×10^3	2.7×10^3	0.6	0.7	0.4	0.25	0.22
Y^{91}	0.57	Bone	7×10^3	57	>500	51	0.0005	0.65	0.55	2.8×10^{-4}	0.14
$Zr^{95} + Nb^{95}$	0.4	Bone	7×10^3	65	104	40	0.0005	—	—	0.1	0.5
Ru^{103}	0.2	Kidney	300	40	20	13.5	<0.0005	—	—	3.5×10^{-3}	0.01
$Ru^{106} + Rh^{106}$	1.4	Kidney	300	365	20	19	$<5 \times 10^{-4}$	0.04	0.04	2.0×10^{-5}	0.01
Te^{127}	0.28	Kidney	300	90.4	15	13	0.25	0.2	0.06	7×10^{-4}	0.02
Te^{129}	0.89	Kidney	300	32	15	10	0.25	0.2	0.06	7×10^{-4}	0.02
I^{131}	0.22	Thyroid	20	8	180	7.7	1.0	0.2	0.2	0.2	0.15
$Cs^{137} + Ba^{137}$	0.57	Muscle	3×10^4	1.2×10^4	17	17	1.0	0.45	0.48	0.48	0.36
$Ba^{140} + La^{140}$	1.06	Bone	7×10^3	12.8	~200	12	0.1	0.96	0.7	0.07	0.2
$Ce^{144} + Pr^{144}$	1.29	Bone	7×10^3	275	500	180	5×10^{-4}	0.8	0.4	2×10^{-4}	0.10
Pm^{147}	0.067	Bone	7×10^3	1.46×10^3	>100	140	5×10^{-4}	0.7	0.35	1.7×10^{-4}	0.09
Np^{239}	0.133	Bone	7×10^3	2.33	100	2.28	0.0005	—	—	5×10^{-4}	0.23

*Table 2. Fraction of total activity of individual isotopes at various times after burst**

Time	Sr^{89}	$Sr^{90} + Y^{90}$	Y^{91}	$Zr^{95} + Nb^{95}$	Ru^{103}	$Ru^{106} + Rh^{106}$	Te^{127}
3.5 hr	2.58×10^{-4}	1.66×10^{-6}	1.75×10^{-5}	2.86×10^{-4}	2.40×10^{-4}	4.15×10^{-6}	2.95×10^{-5}
12 hr	7.35×10^{-4}	4.85×10^{-6}	4.49×10^{-4}	8.09×10^{-4}	6.98×10^{-4}	1.18×10^{-5}	2.13×10^{-4}
1 day	1.36×10^{-3}	9.10×10^{-6}	1.20×10^{-3}	1.53×10^{-3}	1.25×10^{-3}	2.16×10^{-5}	5.23×10^{-4}
2 days	2.70×10^{-3}	1.80×10^{-5}	2.83×10^{-3}	3.01×10^{-3}	2.47×10^{-3}	4.34×10^{-5}	1.05×10^{-3}
4 days	5.17×10^{-3}	3.50×10^{-5}	5.57×10^{-3}	5.97×10^{-3}	4.80×10^{-3}	8.75×10^{-5}	1.03×10^{-3}
7 days	9.70×10^{-3}	6.93×10^{-5}	1.11×10^{-2}	1.11×10^{-2}	9.24×10^{-3}	1.71×10^{-4}	1.76×10^{-3}
14 days	3.20×10^{-2}	2.46×10^{-4}	3.52×10^{-2}	3.61×10^{-2}	2.79×10^{-2}	5.74×10^{-4}	1.80×10^{-3}
28 days	6.40×10^{-2}	6.0×10^{-4}	7.30×10^{-2}	7.70×10^{-2}	5.50×10^{-2}	1.40×10^{-3}	6.60×10^{-4}
105 days	1.10×10^{-1}	2.70×10^{-3}	1.30×10^{-1}	1.50×10^{-1}	6.90×10^{-2}	5.40×10^{-3}	8.30×10^{-4}
210 days	7.0×10^{-2}	7.30×10^{-3}	9.80×10^{-2}	1.40×10^{-1}	3.20×10^{-2}	1.20×10^{-2}	1.0×10^{-3}
365 days	2.60×10^{-2}	1.90×10^{-2}	4.0×10^{-2}	6.90×10^{-2}	7.10×10^{-3}	2.40×10^{-2}	8.0×10^{-4}

Time	Te^{129}	I^{131}	$Cs^{137} + Ba^{137}$	$Ba^{140} + La^{140}$	$Ce^{144} + Pr^{144}$	Pm^{147}	Np^{239}
3.5 hr	7.38×10^{-6}	1.01×10^{-3}	1.29×10^{-6}	1.38×10^{-3}	5.53×10^{-5}	5.35×10^{-8}	7.93×10^{-2}
12 hr	4.26×10^{-5}	3.0×10^{-3}	3.75×10^{-6}	3.97×10^{-3}	1.62×10^{-4}	5.15×10^{-7}	2.64×10^{-1}
1 day	9.10×10^{-5}	5.57×10^{-3}	6.80×10^{-6}	7.38×10^{-3}	2.90×10^{-4}	1.88×10^{-6}	4.05×10^{-1}
2 days	1.78×10^{-4}	1.05×10^{-2}	1.37×10^{-5}	1.37×10^{-2}	5.94×10^{-4}	7.30×10^{-6}	5.43×10^{-1}
4 days	3.38×10^{-4}	1.83×10^{-2}	2.70×10^{-5}	2.35×10^{-2}	1.15×10^{-3}	2.70×10^{-5}	6.0×10^{-1}
7 days	6.0×10^{-4}	2.86×10^{-2}	5.54×10^{-5}	4.16×10^{-2}	2.31×10^{-3}	8.80×10^{-5}	5.12×10^{-1}
14 days	1.89×10^{-3}	5.50×10^{-2}	1.89×10^{-4}	9.84×10^{-2}	7.71×10^{-3}	4.84×10^{-4}	2.28×10^{-1}
28 days	3.50×10^{-3}	4.0×10^{-2}	4.60×10^{-4}	1.10×10^{-1}	1.80×10^{-2}	1.70×10^{-3}	9.0×10^{-3}
105 days	2.90×10^{-3}	2.30×10^{-4}	2.0×10^{-3}	7.80×10^{-3}	6.90×10^{-2}	8.80×10^{-3}	—
210 days	8.0×10^{-4}	6.80×10^{-8}	5.40×10^{-3}	7.30×10^{-5}	1.40×10^{-1}	2.30×10^{-2}	—
365 days	8.0×10^{-5}	—	1.40×10^{-2}	—	2.60×10^{-1}	5.60×10^{-2}	—

* Fractions of total activity of the radioisotopes were determined from data of HUNTER and BALLOU[5] for the thirteen most hazardous fission products and from data calculated for the Np^{239} decay rate from the peak value at 4 days (60 per cent) reported by STEWART *et al.*[6]

in the critical organs with the exceptions of the bone seekers. For this latter case, a correction for non-homogeneous deposition is included in the factor, N, considered to be 5 as recommended by the ICRP[7]. It is not clear whether the value 5 was chosen to account only for the deposition correction or to account also for the sensitivity to radiation of the portion of bone involved. According to MORGAN[8] the factor N includes nonuniform distribution, radiosensitivity and essentialness of body organ. It is believed that the distribution factor should be kept separate from the sensitivity factor. When the distribution factors are better known, a more accurate value of the dose may be calculated.

Solubility

Since there is incomplete data on the solubility of all radioactive compounds in fallout, the MPC's in this report were calculated on the assumption that all radioactive compounds in the mixture are soluble; therefore, the values will be high if appreciable amounts of insoluble radioactive compounds are found in the mixture. As more information becomes available, these values should be modified to correct for various compounds of each radioisotope.

Biological elimination

The use of an effective half-life implies that the biological elimination of radioactive material from the body follows a simple exponential elimination rate. As pointed out by MORGAN and FORD[3] and in Handbook 52[2], this elimination does not follow a simple pattern. However, MORGAN and FORD state that "in most cases this method of determining the maximum permissible concentration in air and water is satisfactory" and that "More refinement in the method of calculating these values will hardly be justified until more detailed biological information and data of greater accuracy are available".

Tissue sensitivity

The calculations were based on the assumption that the critical organs or portions thereof have equal sensitivities to radiation. It is known that some body

tissues are more radiosensitive than others. For example, lymphatic tissue and bone marrow are more sensitive than muscle, and certain parts of the bone are more sensitive than other parts. However, quantitative data on the sensitivity of tissues is not available, although MORGAN[8] indicates that the thyroid sensitivity to iodine-131 is a possible exception and that the MPC for iodine should be doubled. In any case, equations for MPC's should be modified to include a sensitivity factor when values for such factors become available.

Dose criteria

The dose criteria used in this report are 15 rems in 90 days, 150 rems in 30 days. The value of 15 rems in 90 days is based on one of the operational clearance values for special situations recommended by U.S. Naval Radiological Defense Laboratory[9] in 1954 for external whole-body exposure. The recommendation reads: "that individuals will not be exposed to more than 15 rems in a single or accumulated dose in a period of three months and not more than 30 rems in 2 years".

The value of 150 rems in 30 days is based on the medical effects table in the Radiological Recovery Manual[10] for external γ-exposure.

These two military dose criteria are based on the assumption that internal exposure affects the body in the same manner as whole-body exposure. This assumption would probably be true in the case of isotopes, such as sodium, that are deposited uniformly throughout the body, but it would not be true in cases where isotopes are deposited only in certain organs. Assuming equal sensitivity of organs, the assumption implied by the military dose criteria would give low (safe) values. As in the case of biological elimination, more refinement can hardly be justified until additional biological data are available on partial-body effects and radiosensitivity of tissue.

It is pointed out that although the calculated doses are based on the absorption of energy for specific periods of 30 and 90 days, the body continues to be exposed until all radioactive isotopes deposited during the intake period which decay away or are excreted. Therefore, the total lifetime doses will be greater than those reported and may lead to injury at a later time.

RESULTS

The results of this study are presented for: (1) exposures to individual radioisotopes by ingestion and inhalation, and (2) exposures to mixed radioactive fallout by ingestion and inhalation.

Single exposures ($n = 1$) to individual radioisotopes

The radioisotopes with the lowest MPC values required to deliver 15 rems in 90 days for a single ingestion exposure ($n = 1$) are iodine-131 (5.5×10^{-3} μc/cm^3) and strontium-90 plus yttrium-90 (8.0×10^{-3} μc/cm^3). The magnitude of the MPC's of the other radioisotopes considered ranges from about 3 to 100 times that of strontium-90 plus yttrium-90 (Table 3). For a single inhalation exposure, the two radioisotopes that possess the lowest corresponding MPC requirements are also iodine-131 (8.0×10^{-7} μc/cm^3) and strontium-90 plus yttrium-90 (1.0×10^{-6} μc/cm^3). The magnitude of the MPC's of the twelve other radioisotopes increases gradually from 2.4×10^{-6} μc/cm^3 for cerium-144 plus praseodymium-144 to 2.0×10^{-4} μc/cm^3 for neptunium-239 (Table 3). In general, when $n = 1$, the most damaging radioisotopes range in MPC from 5.5×10^{-3} to 5.7×10^{-2} μc/cm^3 for ingestion and from 8×10^{-7} to 5.6×10^{-6} μc/cm^3 for inhalation.

The radioisotope with the lowest MPC required to deliver by ingestion 150 rems in 30 days is iodine-131 (5.9×10^{-2} μc/cm^3). The other radioisotopes have MPC's that increase gradually from 0.25 μc/cm^3 for strontium-90 + yttrium-90 to 8.5×10^{-3} μc/cm^3 for neptunium-239. For inhalation, the corresponding MPC's range from 8.6×10^{-6} μc/cm^3 for iodine-131 to 2.0×10^{-3} μc/cm^3 for neptunium-239 (Table 4).

Various periods of exposure to individual radioisotopes

Examination of Tables 3 and 4 shows that, when n-values vary from 1 to 30 days, the relative magnitudes of the MPC's of the radioisotopes required to deliver 15 rems in 90 days and also 150 rems in 30 days may differ appreciably. These differences reflect differences in the effective half-lives of the radioisotopes.

In summary, these computations have established that: (1) within the first month after fission the mixed fallout material is a hazard mainly because of the following constituents: iodine-131, neptunium-239, strontium-89, zirconium-95 + niobium-95, yttrium-91 and

*Table 3. Maximum permissible concentration ($\mu c/cm^3$) in water and air of radioisotopes of the gross fallout mixture required to deliver to the critical organs** 15 rems *in* 90 days *for seven exposure times (n)*

Isotope	Exposure (days)	Water		Air		Isotope	Water		Air	
		T	G.I.	T	G.I.		T	G.I.	T	G.I.
	1	2.6×10^{-2}	0.25	3.2×10^{-6}	4.5×10^{-5}		$5.5 \times 10^{-3} > 5.5 \times 10^{-3}$		$8.0 \times 10^{-7} > 8.0 \times 10^{-7}$	
	7	3.9×10^{-3}	4.0×10^{-2}	4.9×10^{-7}	7.2×10^{-6}		$1.0 \times 10^{-3} > 1.0 \times 10^{-3}$		$1.5 \times 10^{-7} > 1.5 \times 10^{-7}$	
	14	2.2×10^{-3}	2.3×10^{-2}	2.7×10^{-7}	4.2×10^{-6}		$6.4 \times 10^{-4} > 6.4 \times 10^{-4}$		$9.5 \times 10^{-8} > 9.5 \times 10^{-8}$	
Sr^{89}	21	1.5×10^{-3}	1.5×10^{-2}	1.9×10^{-7}	2.7×10^{-6}	I^{131}	$5.4 \times 10^{-4} > 5.4 \times 10^{-4}$		$7.9 \times 10^{-8} > 7.9 \times 10^{-8}$	
	30	1.2×10^{-3}	1.2×10^{-2}	1.4×10^{-7}	2.2×10^{-6}		$5.0 \times 10^{-4} > 5.0 \times 10^{-4}$		$7.2 \times 10^{-8} > 7.2 \times 10^{-8}$	
	60	7.7×10^{-4}	6.9×10^{-3}	9.1×10^{-8}	1.2×10^{-6}		$4.5 \times 10^{-4} > 4.5 \times 10^{-4}$		$6.7 \times 10^{-8} > 6.7 \times 10^{-8}$	
	90	7.0×10^{-4}	5.4×10^{-3}	8.8×10^{-8}	9.7×10^{-7}		$4.5 \times 10^{-4} > 4.5 \times 10^{-4}$		$6.7 \times 10^{-8} > 6.7 \times 10^{-8}$	
	1	8.0×10^{-3}	9.6×10^{-2}	1.0×10^{-6}	1.7×10^{-5}		0.61	> 0.61	$9.0 \times 10^{-5} > 9.0 \times 10^{-5}$	
	7	1.2×10^{-3}	1.3×10^{-2}	1.5×10^{-7}	2.3×10^{-6}		$8.7 \times 10^{-2} > 8.7 \times 10^{-2}$		$1.3 \times 10^{-5} > 1.3 \times 10^{-5}$	
Sr^{90}	14	6.4×10^{-4}	6.8×10^{-3}	7.1×10^{-8}	1.2×10^{-6}	Cs^{137}	$4.4 \times 10^{-2} > 4.4 \times 10^{-2}$		$6.4 \times 10^{-6} > 6.4 \times 10^{-6}$	
+	21	4.5×10^{-4}	4.8×10^{-3}	5.6×10^{-8}	8.3×10^{-7}	+	$2.9 \times 10^{-2} > 2.9 \times 10^{-2}$		$4.3 \times 10^{-6} > 4.3 \times 10^{-6}$	
Y^{90}	30	3.3×10^{-4}	3.2×10^{-3}	4.2×10^{-8}	5.7×10^{-7}	Ba^{137}	$2.1 \times 10^{-2} > 2.1 \times 10^{-2}$		$3.0 \times 10^{-6} > 3.0 \times 10^{-6}$	
	60	2.1×10^{-4}	1.6×10^{-3}	2.6×10^{-8}	2.9×10^{-7}		$1.1 \times 10^{-2} > 1.1 \times 10^{-2}$		$1.6 \times 10^{-6} > 1.6 \times 10^{-6}$	
	90	1.5×10^{-4}	1.1×10^{-3}	2.3×10^{-8}	2.0×10^{-7}		$8.9 \times 10^{-3} > 8.9 \times 10^{-3}$		$1.3 \times 10^{-6} > 1.3 \times 10^{-6}$	
	1	22.5	9.6×10^{-2}	5.0×10^{-6}	1.7×10^{-5}		0.15	5.7×10^{-2}	5.6×10^{-6}	1.0×10^{-5}
	7	3.4	1.6×10^{-2}	7.3×10^{-7}	2.9×10^{-6}		2.4×10^{-2}	9.3×10^{-3}	9.2×10^{-7}	1.7×10^{-6}
	14	1.8	7.4×10^{-3}	3.9×10^{-7}	1.3×10^{-6}	Ba^{140}	1.5×10^{-2}	5.7×10^{-3}	5.6×10^{-7}	1.0×10^{-6}
Y^{91}	21	1.3	5.2×10^{-3}	2.9×10^{-7}	9.4×10^{-7}	+	1.1×10^{-2}	4.4×10^{-3}	4.3×10^{-7}	7.9×10^{-7}
	30	1.0	3.8×10^{-3}	2.1×10^{-7}	6.8×10^{-7}	La^{140}	9.6×10^{-3}	3.7×10^{-3}	3.7×10^{-7}	6.7×10^{-7}
	60	0.7	2.2×10^{-3}	1.4×10^{-7}	4.0×10^{-7}		8.0×10^{-3}	3.1×10^{-3}	3.1×10^{-7}	5.6×10^{-7}
	90	0.6	1.7×10^{-3}	1.3×10^{-7}	3.1×10^{-7}		7.7×10^{-3}	3.0×10^{-3}	3.0×10^{-7}	5.4×10^{-7}
	1	0.10	0.14	2.5×10^{-6}	2.5×10^{-5}		9.5	4.1×10^{-2}	2.4×10^{-6}	7.4×10^{-6}
	7	1.6×10^{-2}	2.1×10^{-2}	3.5×10^{-7}	3.8×10^{-6}		1.3	5.4×10^{-3}	2.9×10^{-7}	9.7×10^{-7}
Zr^{95}	14	8.1×10^{-3}	1.1×10^{-2}	1.8×10^{-7}	2.0×10^{-6}	Ce^{144}	0.71	2.8×10^{-3}	1.6×10^{-7}	5.0×10^{-7}
+	21	5.4×10^{-3}	7.1×10^{-3}	1.2×10^{-7}	1.3×10^{-6}	+	0.50	1.9×10^{-3}	1.1×10^{-7}	3.4×10^{-7}
Nb^{95}	30	4.4×10^{-3}	5.4×10^{-3}	9.5×10^{-8}	9.7×10^{-7}	Pr^{144}	0.36	1.3×10^{-3}	7.8×10^{-8}	2.3×10^{-7}
	60	2.9×10^{-3}	3.2×10^{-3}	6.3×10^{-8}	5.8×10^{-7}		0.24	6.8×10^{-4}	5.3×10^{-8}	1.2×10^{-7}
	90	2.5×10^{-3}	2.4×10^{-3}	5.5×10^{-8}	4.3×10^{-7}		0.21	4.7×10^{-4}	4.7×10^{-8}	8.5×10^{-8}
	1	2.9	0.28	1.1×10^{-4}	5.0×10^{-5}		54.1	3.9×10^{-2}	1.2×10^{-5}	7.0×10^{-6}
	7	0.45	4.2×10^{-2}	1.7×10^{-5}	7.6×10^{-6}		7.8	5.4×10^{-3}	1.7×10^{-6}	9.7×10^{-7}
	14	0.24	2.2×10^{-2}	9.1×10^{-6}	4.0×10^{-6}	Ru^{106}	4.0	2.8×10^{-3}	8.7×10^{-7}	5.0×10^{-7}
Ru^{103}	21	0.17	1.6×10^{-2}	6.4×10^{-6}	2.9×10^{-6}	+	2.8	2.0×10^{-3}	6.2×10^{-7}	3.6×10^{-7}
	30	0.12	1.2×10^{-2}	4.8×10^{-6}	2.2×10^{-6}	Rh^{106}	2.0	1.4×10^{-3}	4.4×10^{-7}	2.5×10^{-7}
	60	8.3×10^{-2}	7.4×10^{-3}	3.2×10^{-6}	1.3×10^{-6}		1.1	7.1×10^{-4}	2.5×10^{-7}	1.3×10^{-7}
	90	7.7×10^{-2}	6.1×10^{-3}	3.0×10^{-6}	1.1×10^{-6}		0.93	4.8×10^{-4}	2.1×10^{-7}	8.6×10^{-8}
	1	10.9	0.26	4.2×10^{-5}	4.6×10^{-5}		2.3×10^{2}	0.80	4.7×10^{-5}	1.4×10^{-4}
	7	1.6	3.8×10^{-2}	6.2×10^{-6}	6.8×10^{-6}		33	0.11	6.7×10^{-6}	2.0×10^{-5}
	14	0.82	1.9×10^{-2}	3.2×10^{-6}	3.5×10^{-6}		17	5.4×10^{-2}	3.3×10^{-6}	9.7×10^{-6}
Te^{127}	21	0.56	1.3×10^{-2}	2.2×10^{-6}	2.4×10^{-6}	Pm^{147}	11.8	3.6×10^{-2}	2.5×10^{-6}	6.5×10^{-6}
	30	0.41	9.6×10^{-3}	1.6×10^{-6}	1.7×10^{-6}		8.6	2.5×10^{-2}	1.8×10^{-6}	4.5×10^{-6}
	60	0.24	5.3×10^{-3}	9.1×10^{-7}	9.6×10^{-7}		5.3	1.3×10^{-2}	1.1×10^{-6}	2.3×10^{-6}
	90	0.20	3.9×10^{-3}	7.6×10^{-7}	7.1×10^{-7}		4.6	8.4×10^{-3}	9.7×10^{-7}	1.5×10^{-6}
	1	4.4	8.1×10^{-2}	1.7×10^{-5}	1.5×10^{-5}		8.5×10^{2}	0.47	2.0×10^{-4}	8.4×10^{-5}
	7	0.68	1.3×10^{-2}	2.6×10^{-6}	2.4×10^{-6}		2.5×10^{2}	0.14	6.0×10^{-5}	2.5×10^{-5}
	14	0.37	7.1×10^{-3}	1.4×10^{-6}	1.3×10^{-6}		2.2×10^{2}	0.12	5.3×10^{-5}	2.2×10^{-5}
Te^{129}	21	0.26	5.1×10^{-3}	1.0×10^{-6}	9.2×10^{-7}	Np^{239}	2.2×10^{2}	0.12	5.2×10^{-5}	2.2×10^{-5}
	30	0.20	3.8×10^{-3}	7.7×10^{-7}	6.8×10^{-7}		2.2×10^{2}	0.12	5.2×10^{-5}	2.2×10^{-5}
	60	0.13	2.6×10^{-3}	5.1×10^{-7}	4.7×10^{-7}		2.2×10^{2}	0.12	5.2×10^{-5}	2.2×10^{-5}
	90	0.12	2.1×10^{-3}	4.6×10^{-7}	3.8×10^{-7}		2.2×10^{2}	0.12	5.2×10^{-5}	2.2×10^{-5}

* T = tissue; G.I. = gastrointestinal tract.

Table 4. Maximum permissible concentration ($\mu c/cm^3$) in water and air of radioisotopes of the gross fallout mixture required to deliver to the critical organs 150 rems in 30 days for five exposure times (n)*

Isotope	n (days)	Water T	Water G.I.	Air T	Air G.I.	Isotope	Water T	Water G.I.	Air T	Air G.I.
	1	0.55	2.5	6.9×10^{-5}	4.5×10^{-4}		$5.9 \times 10^{-2} > 5.9 \times 10^{-2}$		$8.6 \times 10^{-6} > 8.6 \times 10^{-6}$	
	7	8.9×10^{-2}	0.40	1.1×10^{-5}	7.2×10^{-5}		$1.1 \times 10^{-2} > 1.1 \times 10^{-2}$		$1.6 \times 10^{-6} > 1.6 \times 10^{-6}$	
Sr^{89}	14	5.1×10^{-2}	0.23	6.4×10^{-6}	4.2×10^{-5}	I^{131}	$7.3 \times 10^{-3} > 7.3 \times 10^{-3}$		$1.1 \times 10^{-6} > 1.1 \times 10^{-6}$	
	21	4.1×10^{-2}	0.15	5.1×10^{-6}	2.7×10^{-5}		$6.3 \times 10^{-3} > 6.3 \times 10^{-3}$		$9.2 \times 10^{-7} > 9.2 \times 10^{-7}$	
	30	3.7×10^{-2}	0.12	4.6×10^{-6}	2.2×10^{-5}		$6.1 \times 10^{-3} > 6.1 \times 10^{-3}$		$9.0 \times 10^{-7} > 9.0 \times 10^{-7}$	
	1	0.25	0.96	3.1×10^{-5}	1.7×10^{-4}		8.4	> 8.4	$1.2 \times 10^{-3} > 1.2 \times 10^{-3}$	
Sr^{90}	7	3.8×10^{-2}	0.13	4.8×10^{-6}	2.3×10^{-5}	Cs^{137}	1.3	> 1.3	$1.9 \times 10^{-4} > 1.9 \times 10^{-4}$	
+	14	2.3×10^{-2}	6.8×10^{-2}	2.9×10^{-6}	1.2×10^{-5}	+	0.7	> 0.7	$1.0 \times 10^{-4} > 1.0 \times 10^{-4}$	
Y^{90}	21	1.8×10^{-2}	4.8×10^{-2}	2.2×10^{-6}	8.3×10^{-6}	Ba^{137}	0.5	> 0.5	$7.3 \times 10^{-5} > 7.3 \times 10^{-5}$	
	30	1.6×10^{-2}	3.2×10^{-2}	2.0×10^{-6}	5.7×10^{-6}		0.46	> 0.46	$6.7 \times 10^{-5} > 6.7 \times 10^{-5}$	
	1	4.7×10^{2}	0.96	1.0×10^{-4}	1.7×10^{-4}		1.8	0.57	6.8×10^{-5}	1.0×10^{-4}
	7	76	0.16	1.7×10^{-5}	2.9×10^{-5}	Ba^{140}	0.3	9.3×10^{-2}	1.2×10^{-5}	1.7×10^{-5}
Y^{91}	14	45	7.4×10^{-2}	9.8×10^{-6}	1.3×10^{-5}	+	0.2	5.7×10^{-2}	7.7×10^{-6}	1.0×10^{-5}
	21	33	5.2×10^{-2}	7.2×10^{-6}	9.4×10^{-6}	La^{140}	0.16	4.4×10^{-2}	6.2×10^{-6}	7.9×10^{-6}
	30	32	3.8×10^{-2}	6.9×10^{-6}	6.8×10^{-6}		0.15	3.7×10^{-2}	5.8×10^{-6}	6.7×10^{-6}
	1	2.0	1.4	4.4×10^{-5}	2.5×10^{-4}		2.5×10^{2}	0.41	5.5×10^{-5}	7.4×10^{-5}
Zr^{95}	7	0.32	0.21	7.0×10^{-6}	3.8×10^{-5}	Ce^{144}	40	5.4×10^{-2}	8.9×10^{-6}	9.7×10^{-6}
+	14	0.18	0.11	4.0×10^{-6}	2.0×10^{-5}	+	23	2.8×10^{-2}	5.1×10^{-6}	5.0×10^{-6}
Nb^{95}	21	0.14	7.1×10^{-2}	3.2×10^{-6}	1.3×10^{-5}	Pr^{144}	18	1.9×10^{-2}	4.0×10^{-6}	3.4×10^{-6}
	30	0.13	5.4×10^{-2}	2.9×10^{-6}	9.7×10^{-6}		16	1.3×10^{-2}	3.6×10^{-6}	2.3×10^{-6}
	1	37	2.8	1.4×10^{-3}	5.0×10^{-4}		7.8×10^{2}	0.39	1.7×10^{-4}	7.0×10^{-5}
	7	6.0	0.42	2.3×10^{-4}	7.6×10^{-5}	Ru^{106}	1.2×10^{2}	5.4×10^{-2}	2.6×10^{-5}	9.7×10^{-6}
Ru^{103}	14	3.3	0.22	1.1×10^{-4}	4.0×10^{-5}	+	66	2.8×10^{-2}	1.5×10^{-5}	5.0×10^{-6}
	21	2.7	0.16	1.0×10^{-4}	2.9×10^{-5}	Rh^{106}	50	2.0×10^{-2}	1.1×10^{-5}	3.6×10^{-6}
	30	2.3	0.12	8.9×10^{-5}	2.2×10^{-5}		44	1.4×10^{-2}	9.7×10^{-6}	2.5×10^{-6}
	1	1.4×10^{2}	2.6	5.2×10^{-4}	4.6×10^{-4}		5.8×10^{3}	8.0	1.3×10^{-3}	1.4×10^{-3}
	7	21	3.8×10^{-2}	8.0×10^{-5}	6.8×10^{-5}		9.3×10^{2}	1.1	1.9×10^{-4}	2.0×10^{-4}
Te^{127}	14	11	1.9×10^{-2}	4.4×10^{-5}	3.5×10^{-5}	Pm^{147}	5.3×10^{2}	0.54	1.1×10^{-4}	9.7×10^{-5}
	21	8.7	1.3×10^{-2}	3.3×10^{-5}	2.4×10^{-5}		4.1×10^{2}	0.36	8.6×10^{-5}	6.5×10^{-5}
	30	7.6	9.6×10^{-3}	2.9×10^{-5}	1.7×10^{-5}		3.8×10^{2}	0.25	7.7×10^{-5}	4.5×10^{-5}
	1	50.8	0.81	1.9×10^{-4}	1.5×10^{-4}		8.5×10^{3}	4.7	2.0×10^{-3}	8.4×10^{-4}
	7	8.0	0.13	3.1×10^{-5}	2.4×10^{-5}		2.5×10^{3}	1.4	6.0×10^{-5}	2.5×10^{-4}
Te^{129}	14	4.5	7.1×10^{-2}	1.9×10^{-5}	1.3×10^{-5}	Np^{239}	2.2×10^{3}	1.2	5.3×10^{-4}	2.2×10^{-4}
	21	3.5	5.1×10^{-2}	1.5×10^{-5}	9.2×10^{-6}		2.2×10^{3}	1.2	5.2×10^{-4}	2.2×10^{-4}
	30	3.1	3.8×10^{-2}	1.2×10^{-5}	6.8×10^{-6}		2.2×10^{3}	1.2	5.2×10^{-4}	2.2×10^{-4}

* T = tissue; G.I. = gastrointestinal tract.

barium-140 + lanthanum-140; and (2) increasing the *n*-values from 1 to 90 days and the *t*-values from 3.5 hr to 365 days results in a decrease in the MPC's of the radionuclide components with long half-lives.

Exposures to mixed fallout material

Table 5 is a typical table which gives the different MPC's of the fallout mixture required to deliver to the critical organ a dose from each radioisotope ($Sr^{90} + Y^{90}$ in this case) of 15 rems in 90 days based on exposures ranging from 1 day to 90 days and for eleven different entry times, after fission. From values of this type for other important radioisotopes, it has been possible to calculate the MPC of the mixed fallout material required to deliver a dose based on the total radioisotope content of the mixture.

Tables 6 and 7 indicate the numerical differences in MPC values between exposure times (*n* values) and eleven entry times. For entry times up to about 4 days after fission and for a single ingestion exposure, the MPC's

Table 5. Maximum permissible concentrations ($\mu c/cm^3$) of mixed fallout material in water that will yield doses from the respective radioisotopes to the critical organs of 15 rems in 90 days—seven exposure periods (n) at eleven different times after burst*

Time after burst		Strontium-90 + yttrium-90						
		1	7	14	21	30	60	90
3.5 hr	T	4.8×10^3	7.2×10^2	3.9×10^2	2.7×10^2	2.0×10^2	1.3×10^2	90
	G.I.	5.8×10^4	7.8×10^3	4.1×10^3	2.8×10^3	1.9×10^3	1.0×10^3	6.6×10^2
12 hr	T	1.6×10^3	2.5×10^2	1.3×10^2	93	68	43.3	30.9
	G.I.	2.0×10^4	2.7×10^3	1.4×10^3	9.5×10^2	6.7×10^2	3.7×10^2	2.3×10^2
1 day	T	8.8×10^2	1.3×10^2	70	49	36.3	23.1	16.5
	G.I.	1.1×10^4	1.4×10^3	7.5×10^2	5.1×10^2	3.5×10^2	1.9×10^2	1.2×10^2
2 days	T	4.5×10^2	67	35	25	18.3	11.7	8.3
	G.I.	5.5×10^3	7.2×10^2	3.8×10^2	2.8×10^2	2.0×10^2	1.0×10^2	62
4 days	T	2.3×10^2	34	18.3	12.9	9.4	6.0	4.3
	G.I.	2.8×10^3	3.8×10^2	1.9×10^2	1.3×10^2	90	49	31
7 days	T	1.2×10^2	17	9.2	6.5	4.8	3.0	2.2
	G.I.	1.4×10^3	1.9×10^2	97	67	45	24	16
14 days	T	33	4.9	2.6	1.8	1.3	0.85	0.61
	G.I.	3.9×10^2	52	28	1.8	13	6.8	4.5
28 days	T	13	2.0	1.1	0.75	0.55	0.35	0.25
	G.I.	1.6×10^2	21	11	8.0	5.3	0.28	1.8
105 days	T	3.0	0.44	0.24	0.17	0.12	7.8×10^{-2}	5.6×10^{-2}
	G.I.	36	4.8	2.5	1.8	1.3	0.65	0.27
210 days	T	1.1	0.16	8.8×10^{-2}	6.2×10^{-2}	4.5×10^{-2}	2.9×10^{-2}	2.0×10^{-2}
	G.I.	13	1.8	0.93	0.63	0.44	0.23	0.15
365 days	T	0.42	6.3×10^{-2}	3.4×10^{-2}	2.4×10^{-2}	1.7×10^{-2}	1.1×10^{-2}	7.9×10^{-3}
	G.I.	5.1	0.7	0.36	0.24	0.17	9.0×10^{-2}	5.8×10^{-2}

* T = tissue; G.I. = gastrointestinal tract.

required to deliver a dose of 15 rems in 90 days are greater by a factor of approximately 10 than those required to deliver the same dose in 60 days of exposure ($n = 60$ days). For an entry time of 1 year after fission the MPC's are 8.7×10^{-2} and 1.3×10^{-3} $\mu c/cm^3$, respectively, for single and 90-day exposures.

For inhalation exposures to yield 15 rems in 90 days, the MPC's at 3.5 hr after fission range from 3.7×10^{-4} $\mu c/cm^3$ when $n = 1$ day to 3.7×10^{-5} $\mu c/cm^3$ when $n = 30$ days; at 28 days after fission, from 8.2×10^{-6} $\mu c/cm^3$ to 3.6×10^{-7} $\mu c/cm^3$ (Table 6). In general, the MPC's at 105, 210 and 365 days after fission are quite comparable, ranging from 6.4×10^{-6} to 5.7×10^{-6} for single exposure ($n = 1$) and from 1.5×10^{-7} to 1.2×10^{-7} for 90 days exposure ($n = 90$).

For inhalation exposure to yield 150 rems in 30 days, the MPC's at 3.5 hr after fission range from 4.1×10^{-3} $\mu c/cm^3$ when $n = 1$ to 5.4×10^{-4} $\mu c/cm^3$ when $n = 30$ days; at 28 days after fission, from 9.8×10^{-5} $\mu c/cm^3$ to 7.9×10^{-6} $\mu c/cm^3$, respectively (Table 7). For $n = 1$, 7, 14 and 21 days, the MPC's at 105, 210 and 365 days after fission are similar. At these times the principal contributors to the gross fallout material are zirconium-95 + niobium-95, cerium-144 + praseodymium-144, strontium-89, yttrium-91, strontium-90 + yttrium-90, ruthenium-106 + rhodium-106, and ruthenium-103. For single ingestion ($n = 1$ day) to give 150 rems in 30 days, the MPC's range from 27 $\mu c/cm^3$ at 3.5 hr after burst to 0.83 $\mu c/cm^3$ at 28 days and to 1.1 $\mu c/cm^3$ at 1 year. For $n = 30$ days the corresponding MPC's are 3.6, 6.8×10^{-2} and 3.8×10^{-2} $\mu c/cm^3$. The MPC's for 150 rems in 30 days are greater than those for 15 rems in 90 days by factors ranging from

Table 6. Maximum permissible concentration ($\mu c/cm^3$) in water and air of radioactive fallout for seven exposure times (n) at eleven different times after burst to deliver a dose to the critical organ of 15 rems *in* 90 days (t)*

n	Exposure	Time after fission										
		3.5 hr	12 hr	1 day	2 days	4 days	7 days	14 days	28 days	105 days	210 days	365 days
1	Ing.	2.5	0.81	0.48	0.29	0.18	0.13	6.9×10^{-2}	7.1×10^{-2}	0.10	0.10	8.7×10^{-2}
	Inh.	3.7×10^{-4}	1.2×10^{-4}	6.9×10^{-5}	4.0×10^{-5}	2.4×10^{-5}	1.7×10^{-5}	8.2×10^{-6}	6.9×10^{-6}	6.4×10^{-6}	6.0×10^{-6}	5.7×10^{-6}
7	Ing.	0.54	0.18	0.10	5.8×10^{-2}	3.6×10^{-2}	2.4×10^{-2}	1.2×10^{-2}	1.2×10^{-2}	1.5×10^{-2}	1.4×10^{-2}	1.3×10^{-2}
	Inh.	7.7×10^{-5}	2.5×10^{-5}	1.4×10^{-5}	7.4×10^{-6}	4.5×10^{-6}	2.7×10^{-6}	1.2×10^{-6}	2.0×10^{-7}	7.0×10^{-7}	6.3×10^{-7}	6.5×10^{-7}
14	Ing.	0.37	0.12	6.9×10^{-2}	3.9×10^{-2}	2.4×10^{-2}	1.5×10^{-2}	7.6×10^{-3}	7.1×10^{-3}	8.3×10^{-3}	7.7×10^{-3}	6.4×10^{-3}
	Inh.	5.0×10^{-5}	1.8×10^{-5}	9.0×10^{-6}	5.0×10^{-6}	3.0×10^{-6}	1.8×10^{-6}	8.1×10^{-7}	6.1×10^{-7}	4.6×10^{-7}	4.3×10^{-7}	4.0×10^{-7}
21	Ing.	0.32	0.11	5.9×10^{-2}	3.3×10^{-2}	2.0×10^{-2}	1.3×10^{-2}	5.8×10^{-3}	5.4×10^{-3}	5.7×10^{-3}	5.5×10^{-3}	4.5×10^{-3}
	Inh.	4.4×10^{-5}	1.4×10^{-5}	7.7×10^{-6}	4.2×10^{-6}	2.4×10^{-6}	1.5×10^{-6}	6.2×10^{-7}	4.7×10^{-7}	3.4×10^{-7}	3.1×10^{-7}	2.8×10^{-7}
30	Ing.	0.29	9.7×10^{-2}	5.4×10^{-2}	2.6×10^{-2}	1.8×10^{-2}	1.1×10^{-2}	5.3×10^{-3}	4.5×10^{-3}	4.4×10^{-3}	4.1×10^{-3}	3.3×10^{-3}
	Inh.	3.7×10^{-5}	1.2×10^{-5}	6.7×10^{-6}	3.2×10^{-6}	2.1×10^{-6}	1.2×10^{-6}	4.9×10^{-7}	3.6×10^{-7}	2.5×10^{-7}	2.3×10^{-7}	2.0×10^{-7}
60	Ing.	0.25	8.3×10^{-2}	4.6×10^{-2}	2.5×10^{-2}	1.5×10^{-2}	9.8×10^{-3}	4.3×10^{-3}	3.3×10^{-3}	2.7×10^{-3}	2.4×10^{-3}	1.8×10^{-3}
	Inh.	3.2×10^{-5}	1.0×10^{-5}	5.4×10^{-6}	2.9×10^{-6}	1.7×10^{-6}	1.1×10^{-6}	3.7×10^{-7}	2.6×10^{-7}	1.7×10^{-7}	1.5×10^{-7}	1.4×10^{-7}
90	Ing.	0.24	8.0×10^{-2}	4.4×10^{-2}	2.4×10^{-2}	1.4×10^{-2}	9.2×10^{-3}	3.8×10^{-3}	2.9×10^{-3}	2.1×10^{-3}	1.7×10^{-3}	1.3×10^{-3}
	Inh.	3.1×10^{-5}	1.0×10^{-5}	5.4×10^{-6}	2.9×10^{-6}	1.7×10^{-6}	9.7×10^{-7}	3.6×10^{-7}	2.4×10^{-7}	1.5×10^{-7}	1.3×10^{-7}	1.2×10^{-7}

* Ing. = ingestion; Inh. = inhalation.

Table 7. Maximum permissible concentration ($\mu c/cm^3$) in water and air of radioactive fallout for five exposure times (n) at eleven different times after burst to deliver a dose to the critical organ of 150 rems *in* 30 days (t)

n	Exposure	Time after fission										
		3.5 hr	12 hr	1 day	2 days	4 days	7 days	14 days	28 days	105 days	210 days	365 days
1	Ing.	27	8.5	5.1	3.0	2.0	1.4	0.78	0.83	1.2	1.3	1.1
	Inh.	4.1×10^{-3}	1.3×10^{-3}	7.7×10^{-4}	4.5×10^{-4}	2.8×10^{-4}	1.9×10^{-4}	1.0×10^{-4}	9.8×10^{-5}	1.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}
7	Ing.	5.9	1.9	1.1	0.63	0.40	0.26	0.14	0.14	0.18	0.19	0.15
	Inh.	8.6×10^{-4}	2.8×10^{-4}	1.3×10^{-4}	8.0×10^{-5}	5.6×10^{-5}	3.7×10^{-5}	1.8×10^{-5}	1.7×10^{-5}	2.0×10^{-5}	2.0×10^{-5}	1.9×10^{-5}
14	Ing.	4.3	1.4	0.78	0.43	0.27	0.18	8.8×10^{-2}	8.7×10^{-2}	0.12	0.10	7.9×10^{-2}
	Inh.	7.2×10^{-4}	2.3×10^{-4}	1.3×10^{-4}	6.3×10^{-5}	3.8×10^{-5}	2.5×10^{-5}	1.1×10^{-5}	1.0×10^{-5}	1.1×10^{-5}	1.1×10^{-5}	1.0×10^{-5}
21	Ing.	3.7	1.2	0.67	0.38	0.23	0.15	7.3×10^{-2}	6.8×10^{-2}	8.1×10^{-2}	6.6×10^{-2}	5.2×10^{-2}
	Inh.	5.4×10^{-4}	1.8×10^{-4}	9.7×10^{-5}	5.5×10^{-5}	3.3×10^{-5}	2.0×10^{-5}	9.8×10^{-6}	8.6×10^{-6}	8.9×10^{-6}	8.6×10^{-6}	8.7×10^{-6}
30	Ing.	3.6	1.1	0.62	0.34	0.21	0.14	6.8×10^{-2}	6.0×10^{-2}	6.3×10^{-2}	5.1×10^{-2}	3.8×10^{-2}
	Inh.	5.4×10^{-4}	1.7×10^{-4}	8.4×10^{-5}	5.3×10^{-5}	2.8×10^{-5}	2.0×10^{-5}	9.2×10^{-6}	7.9×10^{-6}	7.7×10^{-6}	7.1×10^{-6}	6.4×10^{-6}

* Ing. = ingestion; Inh. = inhalation.

about 11 to 15. It has been found that, for ingestion, nine out of fourteen radioisotopes have the g.i. tract as the critical organ. Hence, for $t = 30$ or 90, the MPC's to deliver a specified dose are the same. For inhalation, the g.i. tract is the critical organ for four radioisotopes, namely, Ru^{103}, $Ru^{106} + Rh^{106}$, Te^{129} and neptunium-239.

These basic considerations account in part for the magnitude of the MPC's presented in Tables 6 and 7.

DISCUSSION AND CONCLUSIONS

This paper considers the fourteen major radionuclides of the fallout mixture. Their relative amounts in the mixture are given in Table 2 but it is conceivable that, under varying situations, the composition may be different owing to fractionation. It is also likely that different weapon designs may alter not only the relative amounts of the component radioisotopes but also the composition of the fallout material.

On the basis of dose delivered to the tissue, it is apparent that the g.i. tract is the critical organ in determining the MPC for a number of radioisotopes.[1] Whether or not this conclusion is valid depends on the biological effect on the g.i. tissue as compared to the effect of equivalent doses on the other critical organs considered. The same reasoning applies, of course, to other critical organs such as thyroid vs. bone vs. g.i. tract. Unfortunately, the relative sensitivity of tissues to given biological effects is not known. MORGAN[8] suggests that, since the thyroid is resistant to radioiodine, the MPC may be increased with safety by a factor of 2.

Based upon a MPD of 0.3 rem/week under continuous exposure, the acceptable or maximum permissible body burden, q, for individual isotopes has been officially designated by the ICRP.[7] It is 0.6 μc for iodine-131; 1.0 μc for barium-140 + lanthanum-140, cerium-144 + praseodymium-144, and strontium-90 + yttrium-90; 2.0 μc for strontium-89; and 3.0 μc for yttrium-91. The MPC's in water to give the afore-mentioned q values range from 8×10^{-7} μc/cm^3 for strontium-90 + yttrium-90 to 4×10^{-2} μc/cm^3 for yttrium-91; in air, from 2×10^{-10} to 9×10^{-9} μc/cm^3, respectively. The MPC in water for iodine-131 is 6×10^{-5} μc/cm^3; in air, 6×10^{-9} μc/cm^3.

Under conditions of local emergency or, indeed, during the operational recovery phase[10] following the detonation of a nuclear weapon, the MPC's based upon emergency dose levels for *single* and *short-term* exposures at various times after burst must be known. As already stated, the emergency dose levels upon which the present MPC calculations are based are 15 rems in 90 days and 150 rems in 30 days. Data are presented for the major individual isotopes of the fallout and, more significantly, for the total mixture of these isotopes.

No data on the MPC's for single exposures have been accorded official status by the ICRP.[7] However, MORGAN and FORD[3] (pp. 32–39) have presented such data for both water and air exposure for a large number of radioisotopes, assuming several possible critical organs. DICK *et al.*[4] give MPC's in air for forty-nine radioisotopes and for mixed fission products at twenty times after fission, assuming six possible critical organs.

Using the data and equations set forth in NBS Handbook 52[2] and methods given by MORGAN and FORD[3] with appropriate modifications, calculations were made of the MPC's that may be taken into the body by ingestion or inhalation over a number of short, emergency exposure periods ranging from 1 to 90 days.

Accurate estimates of MPC's of radioisotopes for large geographical areas or for long periods of time following an emergency are extremely difficult. More definitive data is needed on the kind and relative amounts and effective energies of radioactive fallout materials.

Recent work emphasizes the importance of the g.i. tract as a critical body organ, since it is critical for over one-half of the principal radioisotopes composing the fallout mixture. Consideration of other tissues as well is especially important here for it is well known that when intake of certain isotopes ceases, the g.i. tract as the critical organ may be replaced by other organs.

As previously indicated, the g.i. tract is the

critical organ for many but not all isotopes; however, its criticality varies with the same radioisotope, depending on the type of intake (Tables 3 and 4). Hence, the MPC calculations for the radioactive fallout are based upon the critical organ yielding the smallest MPC for each element.

In general, the principal contributors to the dose to internal body parts during the first seven days after burst are iodine-131, neptunium-239, barium-140 + lanthanum-140, strontium-89, zirconium-95 + niobium-95, and yttrium-91. The principal contributors from 8 days to 105 days after burst are iodine-131, strontium-89, cerium-144 + praseodymium-144, barium-140 + lanthanum-140 and yttrium-91, whereas cerium-144 + praseodymium-144, zirconium-95 + niobium-95, and strontium-90 + yttrium-90 are the principal contributors from 105 days to 365 days after burst.

As previously stated, the values listed in Tables 6 and 7 are based on military dose criteria. However, they may be used directly to obtain MPC values for other doses during the same time periods of 30 and 90 days by simply using the appropriate factor. For example the MPC for a dose of 1.5 rem in 90 days is one-tenth the tabulated value for 15 rems.

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APPENDIX

Development of Maximum Permissible Concentration Equations for Various Exposure Periods

For tissues other than gastrointestinal (g.i.) tract.

A.1. *Single exposure*

According to Morgan and Ford[3] the dose for a single exposure may be represented by:

$$D = [73.9(f/m)\Sigma(\mathrm{RBE})EI_0T] \times [1 - \exp(-0.639t/T)] \quad \text{(A.1)}$$

where

D = dose in rems for a single exposure (rem = RBE × rad)*

$f = f_w$ or f_a = fraction of ingested (f_w) or inhaled (f_a) activity deposited in critical organ

m = mass of critical organ (g)

(RBE) = relative biological effectiveness of the radiation. It is taken as 1 for β and γ

E = effective energy of radiation of both the radioactive isotope and its daughters (MeV)

E for β-particles $= 0.33(1 - Z^{1/2}/43) \times (1 + E_m^{1/2}/4)E_m$

Z = atomic number

E_m = the maximum energy

E for γ-rays $= \{1 - \exp[-(\mu - \sigma_s)X]\}E_\gamma$

$(\mu - \sigma_s)$ = total coefficient of absorption minus Compton scattering

X = effective diameter of the organ (cm)

$I_0 = I_0^a$, or I_0^w = total activity inhaled (I_0^a) or ingested (I_0^w) (μc)

T = effective half-life in days, $T_bT_r/(T_b + T_r)$

* The rad unit in these equations corresponds to an absorbed dose of 100 ergs/g of tissue.

T_b = biological half-life
T_r = radioactive half-life
t = period from beginning of ingestion to any designated time later
MPC = I_0/V
$V = V_a$ or V_w = volume of air (V_a) or water (V_w)

A.2. *Various periods of exposure*

Equation (A.1) can be rewritten as follows:

$$D_1 = CI_0(1 - e^{-\lambda t}) = CI_1(1 - e^{-\lambda t}) \quad \text{(A.2)}$$

where

$\lambda = 0.693/T$
$C = 73.9(f/m)N(\text{RBE})ET$ (where N = distribution factor)
D_1 = dose from first day to tth day.
I_1 = ingested or inhaled activity during first day

$$D_2 = CI_2\{1 - \exp[-\lambda(t-1)]\} \quad \text{(A.3)}$$

where

D_2 = dose from the second day to the tth day (total period = $t - 1$ days)
I_2 = ingested or inhaled activity during second day

$$D_3 = CI_3\{1 - \exp[-\lambda(t-1)]\}$$

$$D_i = CI_i[1 - \exp\{-\lambda[t - (i-1)]\}] \quad \text{(A.4)}$$

$$\text{Total dose} = \Sigma D = \sum_{i=1}^{n} D_i \quad \text{(A.5)}$$

$$I_1 = \frac{I_0}{n}\exp\left[\frac{-0.693(0)}{T_r}\right] \quad \text{(A.6)}$$

$$I_2 = \frac{I_0}{n}\exp\left[\frac{-0.693(1)}{T_r}\right] \quad \text{(A.7)}$$

$$I_i = \frac{I_0}{n}\exp\left[\frac{-0.693(i-1)}{T_r}\right] \quad \text{(A.8)}$$

where T_r = radioactive half-life. Therefore

$$\Sigma D = \sum_{i=1}^{n}\frac{CI_0}{n}\exp\left[\frac{-0.693(i-1)}{T_r}\right] \times \left(1 - \exp\{-\lambda[t-(i-1)]\}\right) \quad \text{(A.9)}$$

$$= \frac{CI_0}{n}\sum_{i=0}^{n-1}\exp\left(\frac{-0.693i}{T_r}\right) \times \left\{1 - \exp\left[\frac{-0.693(t-1)}{T}\right]\right\} \quad \text{(A.10)}$$

Let $C/n = K$, (A.11)

then

$$\frac{\Sigma D}{KI_0} = \frac{1 - \exp(-0.693n/T_r)}{1 - \exp(-0.693/T_r)} - \sum_{i=0}^{n-1}\exp\{-0.693[(T - T_r)i + tT_r]\} \quad \text{(A.12)}$$

$$= \frac{1 - \exp(-0.693n/T_r)}{1 - \exp(-0.693/T_r)} - \exp(-0.693t/T) \times \sum_{i=0}^{n}\exp[-0.693(T - T_r)i/TT_r] \quad \text{(A.13)}$$

$$= \frac{1 - \exp(-693\,n/T_r)}{1 - \exp(-0.693/T_r)} - \exp(-0.693t/T) \times \frac{1 - \exp[-0.693(T - T_r)n/TT_r]}{1 - \exp[-0.693(T - T_r)/TT_r]} \quad \text{(A.14)}$$

$$(\text{MPC}_w)_n = \frac{I_0{}^w}{V_w} = \frac{I_0{}^w}{2200n} \quad \text{(A.15)}$$

$$(\text{MPC}_a)_n = \frac{I_0{}^a}{V_a} = \frac{I_0}{2 \times 10^7 n} \quad \text{(A.16)}$$

A.3. *MPC for single exposure of g.i. tract from ingestion of contaminated water*

MORGAN and FORD[3] have developed the following equations for the dose to the gastrointestinal tract as the result of the ingestion of radioactively contaminated water for one day (single exposure):

$$D = \frac{51.15\,I\Sigma(\text{RBE})E(h_1 - h_0)f_w{}'}{m'G} \quad \text{(A.17)}$$

where

$f_w{}'$ = fraction reaching critical portion of g.i. tract
$f_w{}' = 1 - f_1$ (f_1 is the fraction going from g.i. tract to blood)
h_0 = time of arrival at critical portion of g.i. tract = 13/24 day when $T_r > \frac{1}{2}$ day
h_1 = time of departure from critical portion of g.i. tract = 31/24 day when $T_r > \frac{1}{2}$ day
$(h_1 - h_0)$ is in fraction of a day
m' = mass of material in critical portion of g.i. tract = 150 g when $T_r > \frac{1}{2}$ day

$$G = \frac{0.693(h_1 - h_0)}{[\exp(-0.693h_0/T_r) - \exp(-0.693h_1/T_r)]T_r}$$

$G = 1$ for $T_r \gg 1$

If it is assumed that half of the radiation reaches the intestinal wall, the MPC in water can be

expressed as follows:

$$(\mathrm{MPC}_w^{\mathrm{g.i.}})_1 = \frac{2I_0}{2200}$$

$$= \frac{Dm'G}{51.15\Sigma(\mathrm{RBE})E(h_1 - h_0)f_w'1100} \quad \text{(A.18)}$$

$$= \frac{D}{K1100} \quad \text{(A.19)}$$

where $$K = \frac{51.15\Sigma(\mathrm{RBE})E(h_1 - h_0)f_w'}{m'G}$$

$$(\mathrm{MPC}_w^{\mathrm{g.i.}})_1 = \frac{1.8 \times 10^{-5}m'DG}{(h_1 - h_0)f_w'\Sigma(\mathrm{RBE})E} \quad \text{(A.20)}$$

A.4. *MPC for single exposure of g.i. tract from inhalation of contaminated air*

Morgan and Ford[3] also reported the following equations:

$$(\mathrm{MPC}_a^{\mathrm{g.i.}})_1 = \frac{2200f_w'}{2 \times 10^7 f_a'}(\mathrm{MPC})_w^{\mathrm{g.i.}} \quad \text{(A.21)}$$

$$= \frac{2 \times 10^{-9}m'DG}{(h_1 - h_0)f_a'\Sigma(\mathrm{RBE})E} \quad \text{(A.22)}$$

where

$f_a' = 0.62\ (1 - f_1)$ = the fraction reaching the critical g.i. tissue.

A.5. *Various periods of ingestion of water*

The total dose to any organ may be expressed by

$$\Sigma D = D_1 + D_2 + D_3 + \ldots\ldots - D_n \quad \text{(A.23)}$$

where

D_1 = dose from intake during first day

D_2 = dose from intake during second day

D_n = dose from intake during nth day

If we let

$$K = \frac{51.15\Sigma(\mathrm{RBE})E(h_1 - h_0)f_w'}{m'G} \quad \text{(A.24)}$$

then equation (A.17) may be written

$$D = KI_0$$

and

$$D_1 = KI_1 = \frac{KI_0}{n}e^{-\lambda(0)} = \frac{KI_0}{n} \quad \text{(A.25)}$$

$$D_2 = KI_2 = \frac{KI_0}{n}e^{-\lambda(1)} \quad \text{(A.26)}$$

$$D_3 = KI_3 = \frac{KI_0}{n}e^{-\lambda(2)} \quad \text{(A.27)}$$

$$D_i = KI_i = \frac{KI_0}{n}e^{-\lambda(i-1)} \quad \text{(A.28)}$$

$$\Sigma D = \sum_{i=1}^{n} D_i = \frac{KI_0}{n}\sum_{i=1}^{n}\exp\left[-\lambda(i-1)\right] \quad \text{(A.29)}$$

$$= \frac{KI_0}{n}\sum_{i=0}^{n-1}e^{-\lambda i} \quad \text{(A.30)}$$

$$= \frac{KI_0}{n}\frac{(1 - e^{-\lambda n})}{(1 - e^{-\lambda})} \quad \text{(A.31)}$$

where $\lambda = 0.693/T_r$. The maximum permissible concentration for various intake periods up to n is:

$$(\mathrm{MPC}_w^{\mathrm{g.i.}})_n = \frac{2I_0}{2200n} = \frac{\Sigma D(1 - e^{-\lambda})}{K(1 - e^{-\lambda n})1100} \quad \text{(A.32)}$$

By dividing equation (A.32) by equation (A.19) for $(\mathrm{MPC}_w^{\mathrm{g.i.}})_1$ one obtains:

$$\frac{(\mathrm{MPC}_w^{\mathrm{g.i.}})_n}{(\mathrm{MPC}_w^{\mathrm{g.i.}})_1} = \frac{(1 - e^{-\lambda})}{(1 - e^{-\lambda n})} \quad \text{(A.33)}$$

$$(\mathrm{MPC}_w^{\mathrm{g.i.}})_n = (\mathrm{MPC}_w^{\mathrm{g.i.}})_1\frac{(1 - e^{-\lambda})}{(1 - e^{-\lambda n})} \quad \text{(A.34)}$$

A.6. *Various periods of inhalation of air*

Equation (A.21) was similarly modified to obtain the expression:

$$(\mathrm{MPC}_a^{\mathrm{g.i.}})_n = (\mathrm{MPC}_a^{\mathrm{g.i.}})_1\frac{(1 - e^{-\lambda})}{(1 - e^{-\lambda n})} \quad \text{(A.35)}$$

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MAXIMUM PERMISSIBLE ACTIVITY (MPA) FOR FISSION PRODUCTS IN AIR AND WATER

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Abstract—When mixed fission products are present in air and water, the maximum permissible activity (MPA) is dependent on the relative activity of the individual isotopes. This, in turn, is dependent on the age of the products. The MPA can be determined most accurately from measurements of the isotopic concentrations; however, the separation of mixed fission products into their constituent activities is extremely complicated and costly.

In order to provide a simple method of evaluating the internal radiation hazard, MPA's for an ideal mixture of fission products vs. its age were calculated and plotted. The resulting curves show that the MPA's change appreciably with time. Between 1 day and 3 years for 40 hr per week occupational exposure, they decrease from 2.7×10^{-7} to 1.2×10^{-9} $\mu c/cm^3$ (microcuries per cubic centimeter) for air, and from 1.6×10^{-3} to 1.7×10^{-5} $\mu c/cm^3$ for water. From the curves a first approximation of the MPA's, and therefore the allowable fission product activity in air and water, can be obtained. In order to use the curves, one needs to know only the disintegration rate per unit volume and the approximate age of the products.

INTRODUCTION

LIMITS for unknown mixtures have been applied to fission products, but these are extremely conservative since they are based on the allowable activity of the most hazardous isotope present. To evaluate the hazard of mixed fission product activity properly, the contribution to the total activity and the MPC (maximum permissible concentration) of each isotope must be taken into account. The contribution of each isotope can be determined most accurately from the actual measurements; however, the separation of mixed fission products into their constituent activities is extremely complicated and costly. In order to provide a simplified means of evaluating the hazard from mixed fission products, the maximum permissible activity (MPA) in air and water for the mixture of isotopes resulting from the simultaneous, slow-neutron fission of uranium-235 were calculated. The curves which resulted can be used as a first approximation in evaluating the hazard from fission product mixtures. In applying the curves, it must be borne in mind that fractionation, burn-up, different neutron energies and use of different fissionable materials may alter the actual maximum permissible activity somewhat.

METHOD OF CALCULATION

The curves that are presented in this article as guide lines for evaluating the internal radiation exposure problems are based on the method of calculating permissible concentrations of radioisotopic mixtures recommended by the National Committee on Radiation Protection (NCRP),[1] and on fission product populations by BOLLES and BALLOU[2].

BOLLES and BALLOU published the activities resulting from the simultaneous fission of 10,000 uranium-235 atoms by slow neutrons. Values were presented for two theories, the theory of PRESENT, and the theory of GLENDENIN, CORYELL and EDWARDS. Since there is little difference in the results of the two, the values given by PRESENT's theory were arbitrarily chosen for use in the computations that were performed.

MPC values are given by the NCRP[1] for sixty-two of the isotopes listed by BOLLES and BALLOU as contributing significantly to total activity at times greater than 14.5 min after the production of the fission products. Consideration of only sixty-two isotopes excludes a significant portion of the activity. In order to take explicit account of as large a portion of the fission product activity as practicable, the MPC's for eight additional isotopes were calculated employing the same methods used by the NCRP to determine

Table 1. *Fraction of total activity due to isotopes with known MPC's*

Fraction of total activity due to isotopes with known MPC's	Age	Fraction of total activity due to isotopes with known MPC's	Age
0.0519	14.5 min	0.9976	97.3 days
0.1017	31.2 min	1.000	143 days
0.2129	1.12 hr	1.000	208 days
0.5192	3.52 hr	1.000	1.20 year
0.7923	11.1 hr	0.9925	2.60 years
0.9500	23.8 hr	0.9983	5.58 years
0.9828	2.13 days	0.9990	12.0 years
0.9858	4.57 days	0.9990	25.7 years
0.9926	6.70 days	0.9996	55.3 years
0.9971	14.4 days	0.9996	81.0 years
0.9983	30.9 days		

MPC values.[1] Table 1 shows the fraction of the total activity contributed by the seventy isotopes whose MPC's were known. It should be noted that this was more than 98 per cent of the activity at 2.13 days and times thereafter. The MPC's that were not given by the NCRP and were not calculated are for a number of fission product isotopes which have extremely short half lives or which contribute only a small portion of the total activity at any time during the decay cycle.

MPA's were determined considering twelve possible critical organs. Since the solubility of a radioisotope effects the MPA for the lungs and the gastrointestinal (GI) tract, MPA's were calculated for each of these organs for both the soluble and insoluble states.

The MPA's were calculated from the formula[1]:

$$\sum_{i=1}^{N} \frac{f_i \mathrm{MPA}}{\mathrm{MPC}_i} = 1$$

where MPA is maximum permissible activity in air or water for the age and critical organ considered;

MPC_i is maximum permissible concentration in air or water for the ith isotope;

f_i is the fraction of the "total activity" due to the ith isotope;

N is the number of isotopes present which had known MPC's for the time and organ considered.

The "total activity" includes only the activity from isotopes with known MPC's. Deriving the MPA in the above manner assumes that a unit of activity from the isotopes with known MPC's and a unit of activity from those with unknown MPC's represent equal hazards. This makes the resulting MPA's somewhat conservative (low), especially at early times when most of the isotopes with unknown MPC's have extremely short half lives.

The lowest MPA value of the fission product mixture for any of the organs at a particular time was chosen as the MPA of the mixture at that time. These lowest MPA values are plotted in Fig. 1 for air and Fig. 2 for water. Table 2 shows the most critical organ as a function of the age of the fission debris. A slight amount of conservatism is present in the curves in Figs. 1 and 2 when the most critical organ is the GI tract or the lungs, since the limits are established assuming a completely insoluble contaminant. Most fission product mixtures will undoubtedly consist of both soluble and insoluble material, and the true MPA will be between the MPA for the soluble and insoluble forms.

USE OF CURVES

The MPA's were calculated for a 40-hr work week in a controlled situation. For continuous exposure in the neighborhood of controlled areas, these values should be divided by 30. (A factor of approximately 3 is required to convert to continuous occupational exposure and a factor of 10 to convert from continuous occupational exposure to the limits for the public living in the neighborhood of a controlled area.)[3] For the population at large, a reduction by an additional factor of 3 is required.[3] For controlled exposure differing from 40 hr per week, the following formula should be used:[1]

$$\mathrm{MPA} = \frac{40}{H} (\mathrm{MPA})_{40}$$

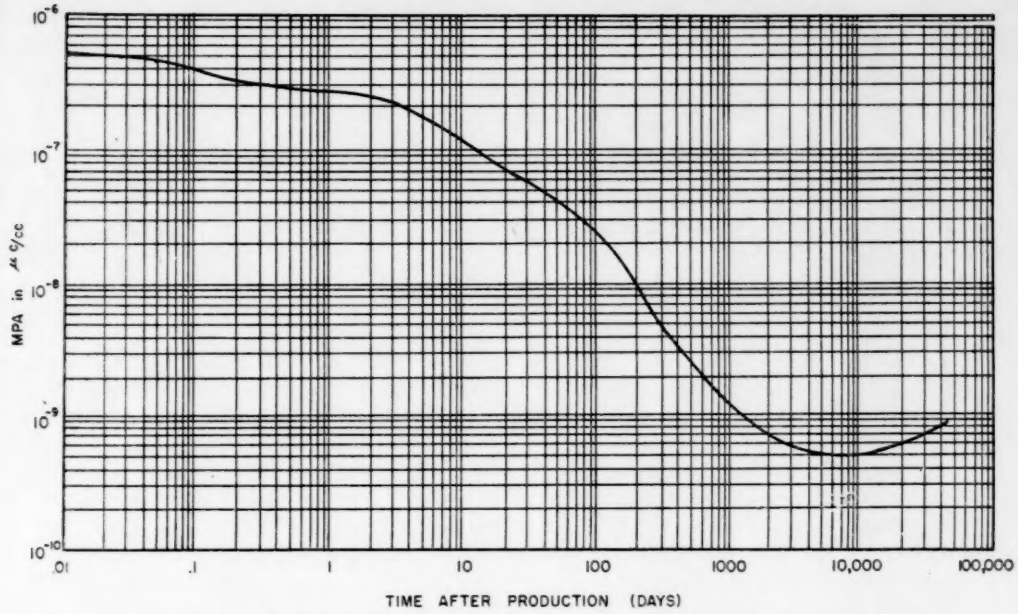

FIG. 1. MPA of fission products in air for 40 hr per week occupational exposure.

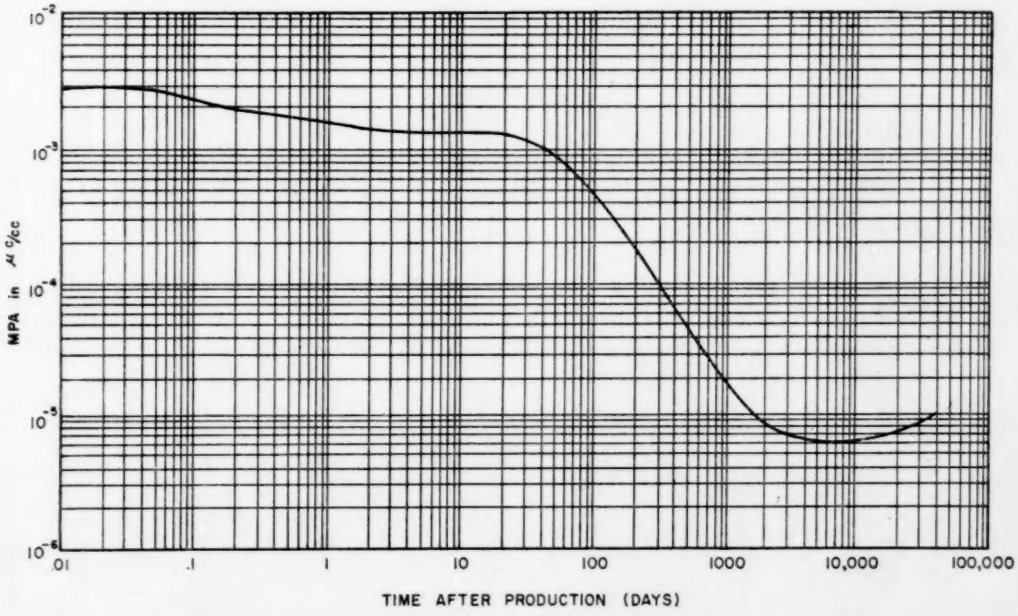

FIG. 2. MPA of fission products in water for 40 hr per week occupational exposure.

Table 2. Most critical organ for the fission product mixture

Air		Water	
Time	Most critical organ	Time	Most critical organ
14.5 min–2.80 days	GI (insoluble)	14.5 min–7.0 days	GI (insoluble)
2.80 days–3.85 days	Thyroid	7.0 days–46.0 days	Thyroid
3.85 days–38.5 days	Lungs (insoluble)	46.0 days–81.0 years	Bone
38.5 days–81.0 years	Bone		

where MPA is the maximum permissible activity to be determined.

$(MPA)_{40}$ is the maximum permissible activity for a 40-hr work week in controlled area, which can be determined from the graphs in Figs. 1 or 2;

H is number of hours of controlled occupational exposure per week.

Once the MPA has been determined from Fig. 1 for air or Fig. 2 for water, it can be compared with the total disintegration rate (β- plus γ-disintegrations) to assess the potential hazard from the radioactivity present. If only the β-disintegration rate is known, it can be used instead of the total disintegration rate without introducing appreciable error. It should be noted that the MPA's are for air and water used by humans. Higher levels are permissible if the activity will be reduced to or below the MPA limit before the air or water is inhaled or ingested. This reduction may be brought about by decay, dilution, settling out of the radioactive isotopes and other factors.

CONCLUSION

In evaluating the hazard from fission product mixtures, the use of limits for unknown mixtures are ultraconservative, and measurement of the activity of each fission product radioisotope to determine the permissible activity is extremely costly in money and time. The curves presented in this article as guide lines in evaluating the internal exposure from fission products are easy to use and provide a realistic basis for estimating the allowable amount of fission products activity that may be inhaled or ingested.

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NOTES

Medium-level Continuous α-Air Monitor*

(*Received* 6 *June* 1960)

Natural background

THE development of α-air monitoring instruments[1] has always been complicated by the natural radioactivity contained in air. This natural radioactivity results from radon and thoron gases emitted by radioactive materials in the earth.

Air drawn through a filter causes a radioactive material build-up on the filter. The half-life for radon is approximately 30 min and for thoron, 10.6 hr; consequently, the activity on the filter increases rapidly at first and then at a slower rate in accordance with the longer half-life and smaller abundance of thoron daughters.[2] The initial build-up of the radon daughters constitutes the principal activity that is collected by the filter since the radon concentration in the atmosphere is generally greater than that of thoron.

Changing atmospheric conditions cause the natural background concentrations to fluctuate. During a temperature inversion which usually occurs in the early morning, the denser cold air near the ground is covered by a layer of warm air. This effect causes the radon and thoron gases to remain close to the

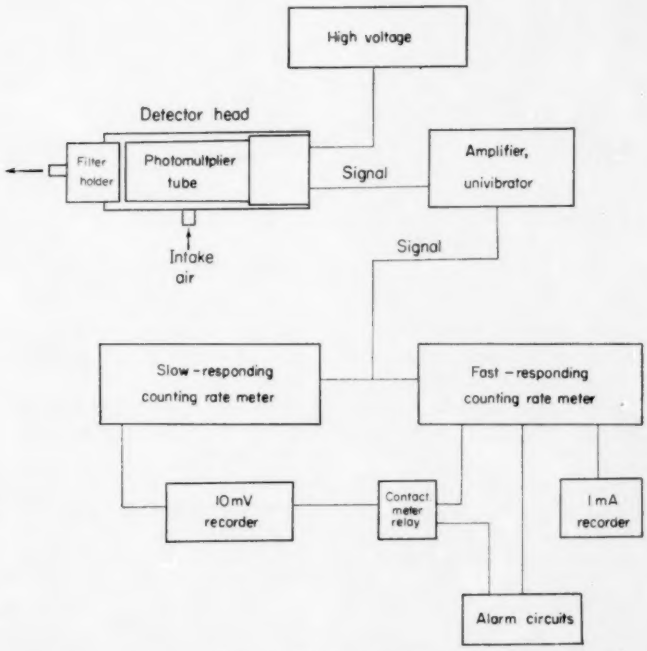


FIG. 1. Block diagram.

* Work performed under Contract No. AT(45-1)-1350 for the U.S. Atomic Energy Commission.

surface of the earth resulting in a high concentration of natural background. In a normal atmosphere with lower warm air, the gases rise and mix with the cold air due to natural convection.

During a series of tests in 1957, a maximum amount of 4000 dis/min of background activity was collected on a molecular membrane filter sampling the air at 3 ft^3/min. The maximum rate of build-up of the filter activity during this inversion was 680 dis/min per hr.

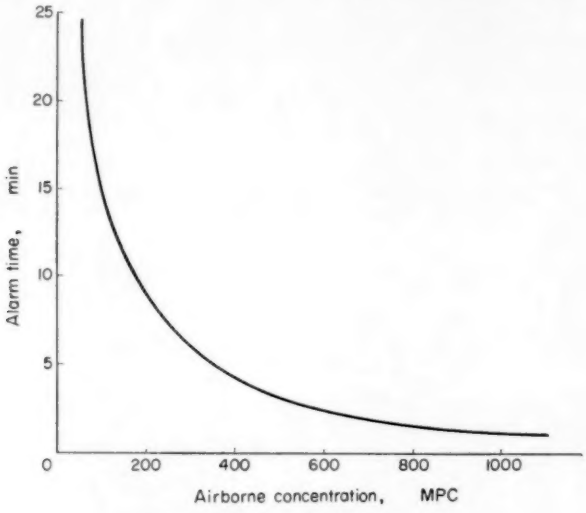


FIG. 2. Differential alarm times.

Instrument characteristics and description

The basic instrument includes a detector head, a high-voltage supply, an amplifier, a univibrator, two counting-rate meters,[3] a 1-mA recorder,[4] a 10-mV recorder,[5] a high–low 20-μA contact meter relay,[6] and alarm circuits. A block diagram of the instrument is shown in Fig. 1.

The detector head includes a molecular membrane filter[7] for collecting particles from the monitored air, a photomultiplier tube[8] with the photocathode window covered with 10 mg/cm^2 of zinc sulfide (silver activated), and a cathode follower circuit.

Operation

Air is drawn through the molecular membrane filter at the rate of 3 ft^3/min. Collected α-emitters cause scintillations in the zinc sulfide, which, in turn, cause electrical pulses to be produced by the phototube. These pulses are amplified and trigger the univibrator which drives both of the counting-rate meters.

The voltage at the recorder output of each of the counting-rate meters varies directly with the meter reading or the indicated counting rate. The high–low 20-μA contact meter relay and the 10-mV recorder are used to monitor the voltage difference between the recorder outputs of the two counting-rate meters.

Modified *RC* circuits are used to average the input signals in the counting-rate meters. The *RC* time of the fast-responding counting-rate meter is approximately $2\frac{1}{2}$ min which is sufficient to reduce fluctuations caused by disintegrations from an α-emitter but allows the instrument to rapidly follow generally increasing or decreasing input signal. The *RC* time of the slow-responding meter is approximately ten minutes. Since the *RC* times of the two meters are different, current flows in the microammeter if the input signal is changing. The amount of current is proportional to the rate of change of the input signal.

The 1-mA recorder is driven by the fast-responding counting-rate meter to produce a record of the total radioactivity deposited on the filter.

Alarms

Three instrument alarm circuits are incorporated for condition indication.

The fixed-alarm circuit in the fast-responding counting-rate meter becomes energized when the total deposition of the α-activity on the filter exceeds 4800 dis/min. This alarm detects the build-up of long half-life α-emitters at slow rates. Laboratory

tests indicated that the natural background activity deposited on a filter during an inversion was always less than 4000 dis/min with a sampling rate of 3 ft^3/min.

The differential alarm circuit energizes when the filter activity build-up, due to α-emitters, exceeds 1260 dis/min per hr. Laboratory tests determined the maximum deposition rate was 672 dis/min per hr during an inversion. The high-level contacts in the contact meter relay energize the alarm circuit. The time required for sufficient current to flow in the microammeter to energize the alarm circuit is approximately inversely proportional to the airborne concentration. Fig. 2 illustrates the relationship between alarm times and various airborne concentrations. Figs. 3 and 4 relate the microammeter current and the alarm times for specific airborne concentrations.

During an inversion, natural background α-emitters are deposited on the filter, and the meter indicates in the positive direction from the midscale zero. When the natural background α-emitters decay after the inversion, the meter indicates in a negative direction; therefore, if there is a sudden increase in airborne Pu^{239} concentration during an inversion, the alarm time will be shorter than if the airborne Pu^{239} concentration increase occurs when the α-emitters from the natural background are decaying.

An added safety feature is provided by the low-limit contact on the contact meter relay. This contact energizes an alarm circuit if the filter ruptures while there is some activity present on the filter.

Calibration and sensitivity

A standard Pu^{239} source is inserted in front of the filter to determine the geometry of the detector.

Different rates of activity build-up on the filter were simulated by driving the frequency control of a pulse generator with a gear and timing motor assembly. The pulse generator was driven at the changing rates of 2.8, 5.25, 7, 9.45 and 94.5 c/s per hr which simulated deposition rates of 672 dis/min per hr (background rate of change, 30 MPC), 1260 dis/min per hr (55 MPC), 1680 dis/min per hr (75 MPC), 2268 dis/min per hr (100 MPC) and 22,680 dis/min per hr (1000 MPC).

It was determined by further experimentation with the pulse generator that the frequency change of 5.25 c/s per hr energized the alarm circuit in 20 min. Assuming that the geometry of the detector arrangement was 25 per cent, this frequency change simulated a deposition rate of 56.7×10^{-5} μc/min. This would be equivalent to an airborne concentration of 55 MPC.

The recorder in parallel with the contact meter relay records the voltage[5] difference between the counting-rate meter outputs. A 1-mA recorder graphs the counting rate of the fast-responding counting-rate meter. The charts of the two recorders

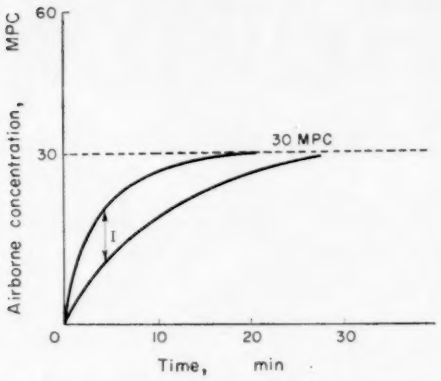


FIG. 3. Averaging times of counting rate meters with an airborne concentration of 30 MPC. (*I* indicates current flow between counting-rate meters.)

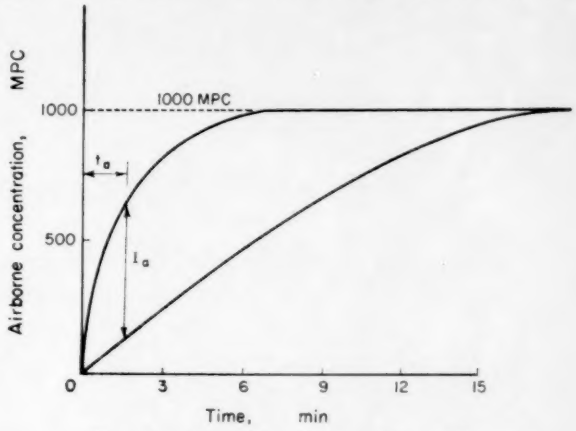
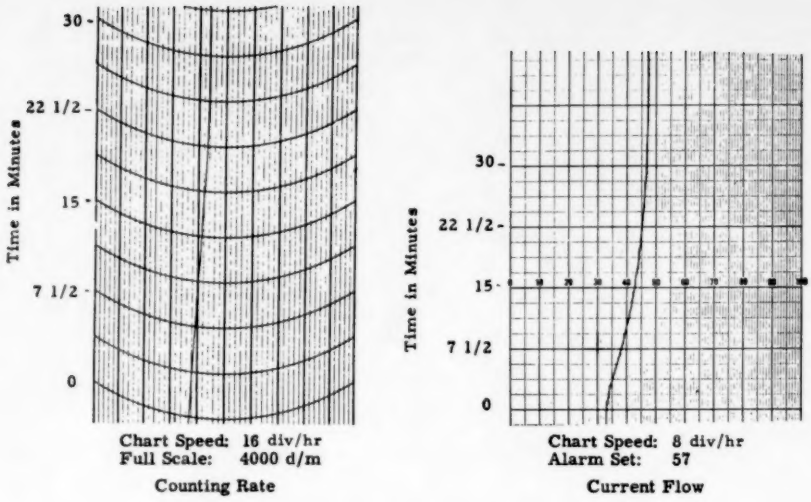


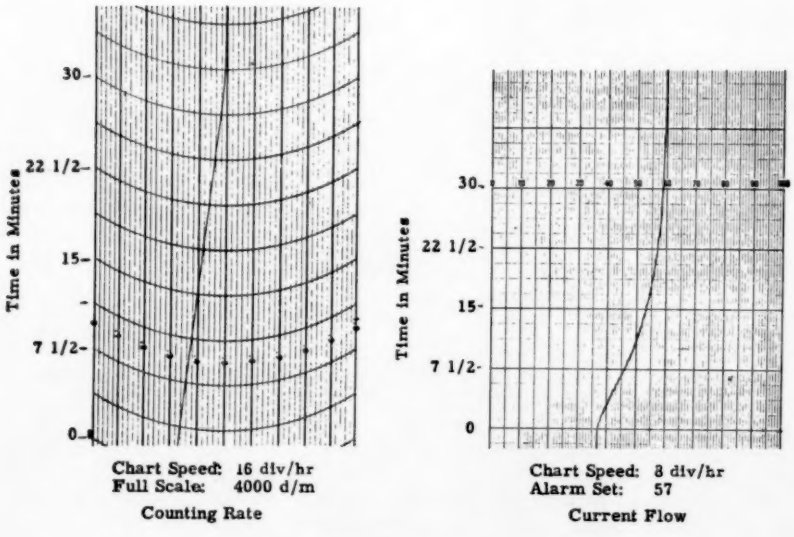
FIG. 4. Averaging times of counting-rate meters with an airborne concentration of 1000 MPC. (I_a indicates sufficient current flow between the counting-rate meters to energize differential alarm circuit; t_a indicates alarm time with this airborne concentration.)



Note: 1) Input Signal Equivalent to a Deposition Rate of 672 d/m/hr or an Airborne Concentration of 30 MPC.

2) This is the Maximum Background Rate of Change Recorded, Therefore System Did Not Alarm.

FIG. 5. Chart of two recorders operating simultaneously.



Note: 1) Input Signal Equivalent to a Deposition Rate of 1260 d/m/hr or Airborne Concentration of 55 MPC.

2) Alarmed in 20 Minutes.

FIG. 6. Chart of two recorders operating simultaneously.

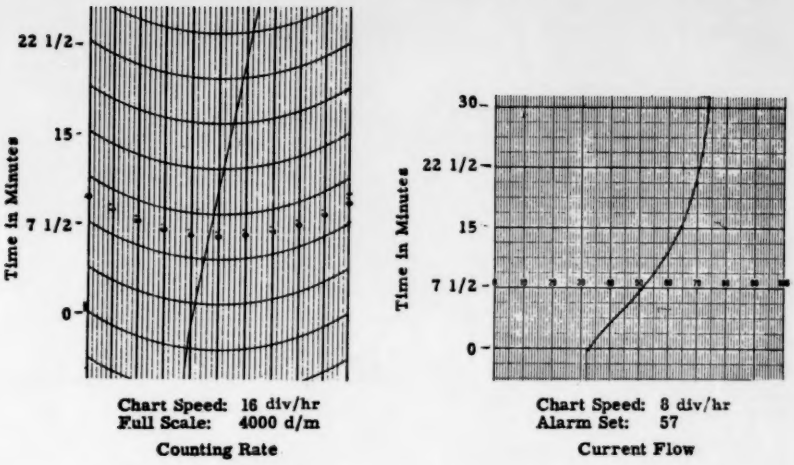


FIG. 7. Chart of two recorders operating simultaneously.

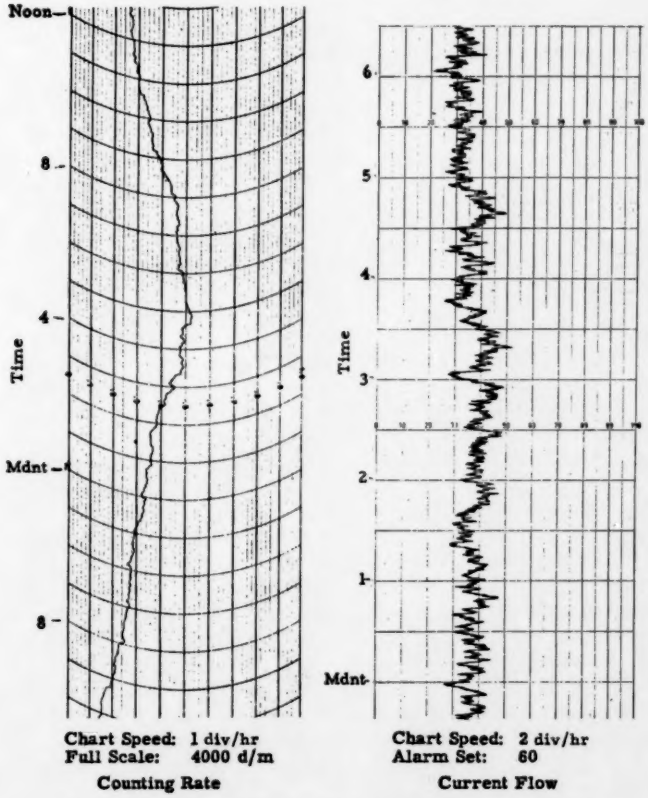


FIG. 8. Deposition of natural background.

Table 1. *Alarm sensitivity table*

Governing alarm	Plutonium concentration (MPC)	Deposition rate of plutonium on filter (dis/min per hr)	Approximate alarm time (min)
Fixed	1	22	12,600
Fixed	10	226	1260
Differential	55	1260	20
Differential	75	1680	18
Differential	100	2268	12
Differential	1000	22,680	1.5

operating simultaneously with various rates of a changing input signal are shown in Figs. 5–7. A normal recorded inversion is shown in Fig. 8. The alarm setting and sensitivity are determined from these charts. Table 1 indicates the approximate alarm times.

The maximum permissible amount, MPA, of insoluble plutonium in the lungs for a standard man is 0.02 μc.[9] Using this figure, and assuming 12 per cent retention in the lungs, a standard man stationed in a room monitored with the medium-level continuous α-air monitor would retain less than tion were greater than 55 MPC, and the man left 0.03 per cent of the MPA if the airborne contamina- the area at the time of the alarm.

Acknowledgments—The writer is indebted to L. V. Zuerner for conducting laboratory tests on the natural radon and thoron background and to G. E. Driver and W. G. Spear for advice concerning the circuitry.

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Richland, Washington

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5. Varian Associates, Inc., Chart Recorder, Model G-11A (10 mV).
6. Assembly Product, Inc., Microampere Contact Meter Relay, Model 451–20.
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The Vibrating-Reed Electrometer as a Low-Level α-Detector*

(*Received* 21 *April* 1960)

Stable low-level α-counters have increased in importance in health physics work as more persons are exposed to α-emitters and the transuranic elements become more abundant for research and industrial applications. Low-level α-counters operating in the proportional region have been the standard detection method in bioassay work for many years, but the nuclear track emulsion technique has been developed[1] for analysis of large numbers of samples on a routine basis with a declared sensitivity limit of less than 0.05 α-disintegrations per minute per sample.

This Note describes a low-level α-counter with the detecting element a vibrating-reed electrometer in the rate of charge mode of operation. Fig. 1 shows succinctly the principle involved. As the electrometer capacitor is charged due to a steady background, discontinuities appear due to α-particle disintegrations in the chamber. In a report in 1947[2] of the vibrating-reed electrometer, these α-discontinuities were noted, but this instrument has normally been used to measure a more continuously emitting source of α- or β-particles. In Fig. 2 is shown the design of an α-chamber that is connected to a commercial vibrating-reed (dynamic condenser) electrometer. The chamber is continuously flushed with argon to eliminate the added background that would arise if air were used as the counting gas. Construction material is stainless steel (SS-304) which has a relatively low α-contamination. The height of the chamber is designed so that an α-particle of 4.5 MeV will dissipate all its energy in the gas if it is deposited approximately in the center of the bottom plate.

* Work assisted in part by U.S. Atomic Energy Commission.

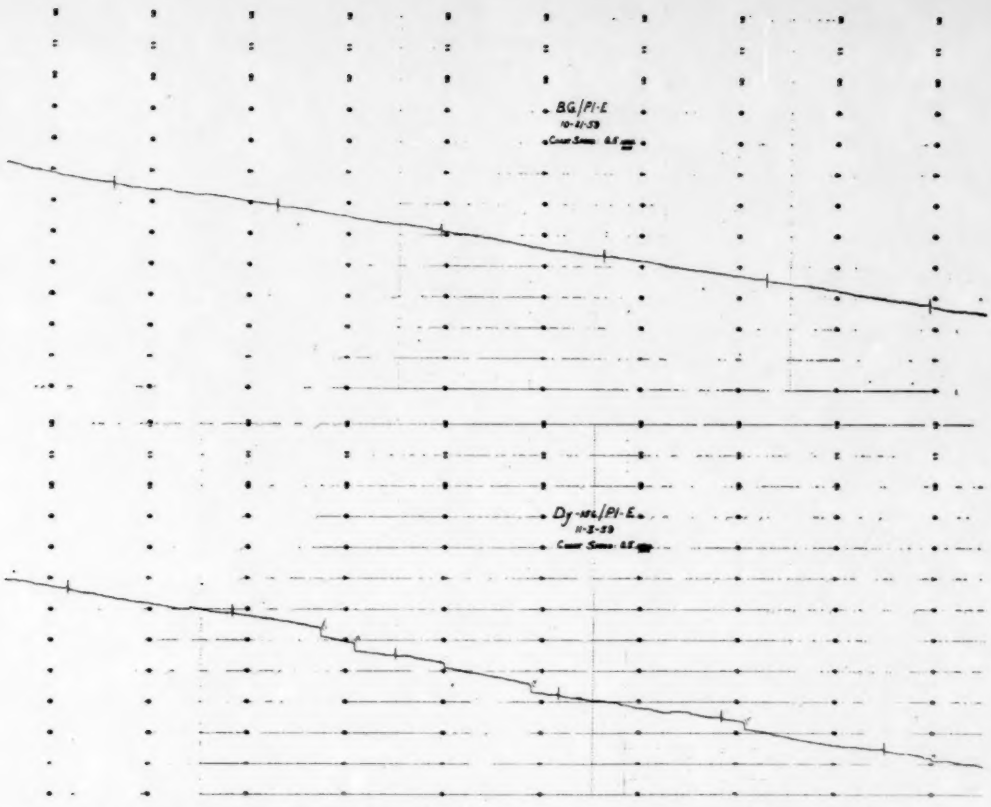

FIG. 1. Steady condenser charge with α-discontinuities superimposed.

The complete counter set-up is shown in Fig. 3; from right to left in the picture is shown the counting gas cylinder, counting chamber with amplifier head, main signal detector and recorder. A voltage of +90 V was used on the center electrode in the chamber. The counting geometry is assumed to be 50 per cent.

The massive shielding and anticoincident array that is characteristic of low-level β- and γ-counting is not required in this method of α-detection. The α-background arises from two sources; one is the instrumental fluctuations that may be mistaken for α-pulses and the other, the natural occurring α-emitters in the chamber construction material.

The α-background obtained with this instrument on one background run was 0.036 dis/min; it may vary between background runs by as much as ±0.004 dis/min. With counting times of 4000–5000 min, it is therefore possible to detect activity of about 0.04 dis/min above background. (But in the interpretation of activity of this magnitude one must take cognizance of the treatment of the sample, since practically any chemistry or grinding of the sample could easily introduce this amount of α-activity.) The maximum activity for which this method is applicable depends on recorder chart speed. For a chart speed of 0.5 in/min, the upper limit is about 4 dis/min or 2 counts/min per sample. This is the level at which coincident counts on the chart become a problem.

Electromagnetically enriched Sm^{147}, a well-known α-emitter of 2.18 MeV, was obtained from Oak Ridge National Laboratory. Thirty-five micrograms of this isotope, which gives approximately 1 dis/min, was deposited as an almost invisible source on a stainless-steel planchet by slurrying the sample in absolute ethyl alcohol and evaporating under an infrared lamp. Fig. 4 gives the pulse-height distribution with the mean equal to pulse-height 7 (arbitrary unit).

As well as providing an additional sensitive and stable method of detection of α-rays in practical

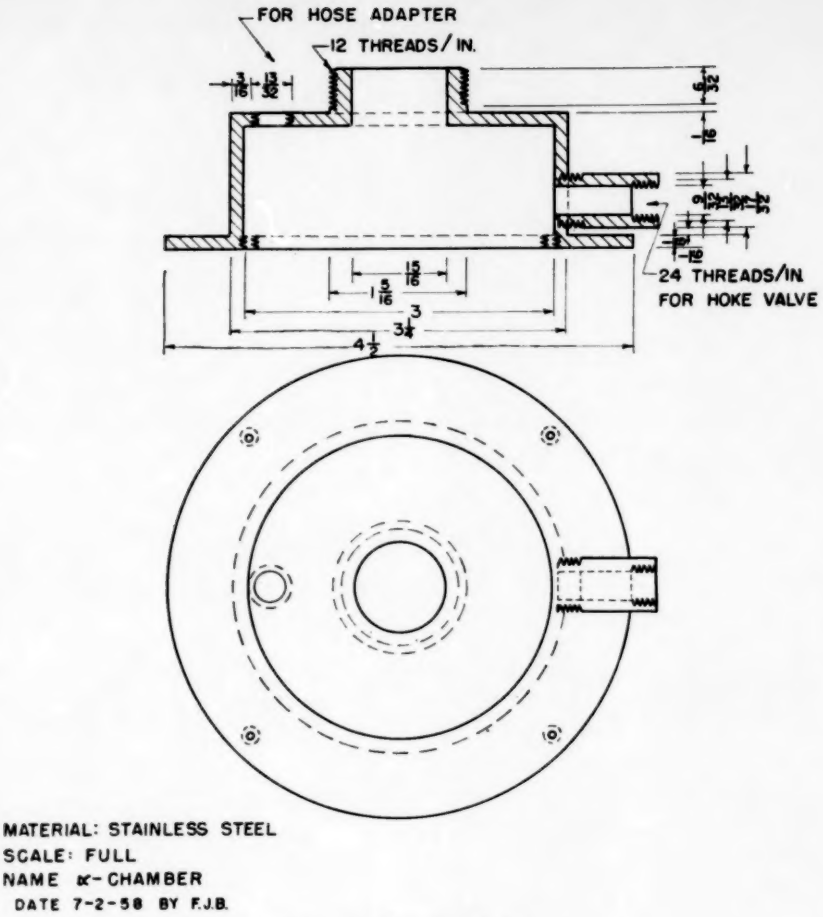

FIG. 2. α-Chamber design.

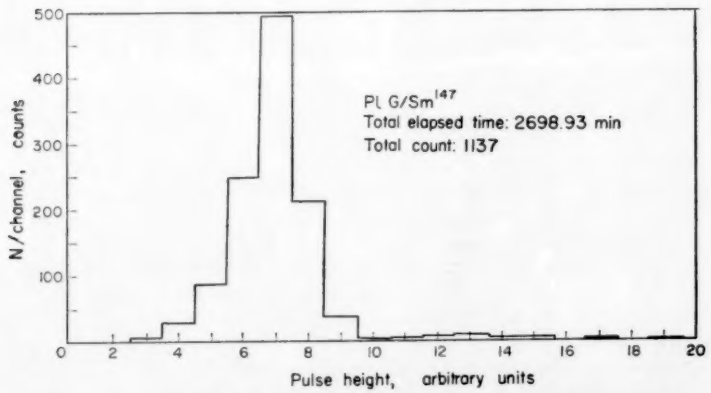

FIG. 4. Pulse-height distribution of α-particles from Sm^{147}.

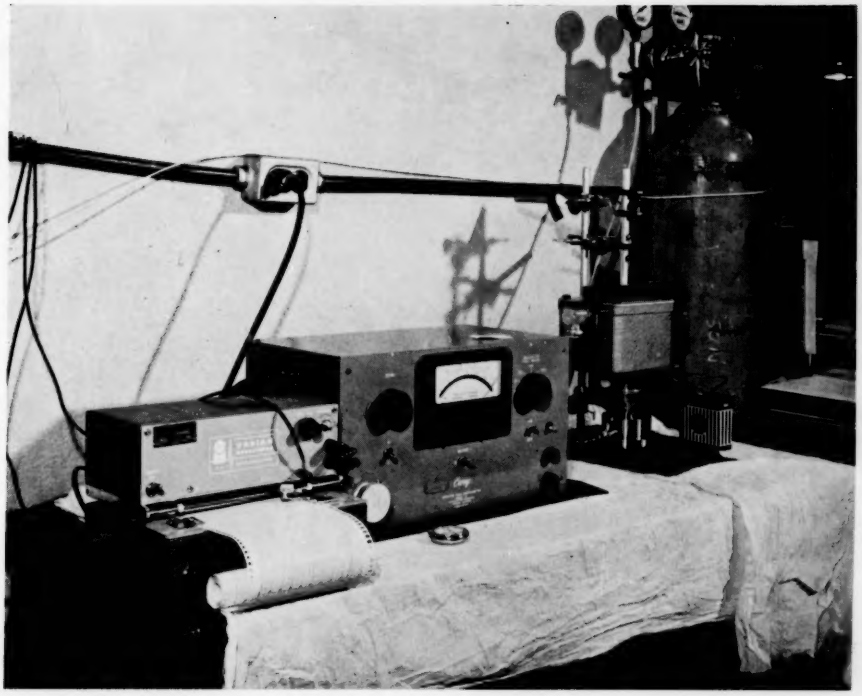

FIG. 3. Complete chamber and electronic set-up.

health physics work, it has also proved to be a useful tool in the investigation of long-life α-emitters.[3]

Acknowledgements—The author would like to thank Prof. M. H. KURBATOV and Prof. J. D. KURBATOV for their encouragement and support during this work.

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An Integrating Ion Chamber Dosemeter

(*Received* 13 *June* 1960)

IN A recent paper,[1] HANDLOSER described integrating ion chamber instruments which have been found useful for radiation survey work on the Cosmotron. There may be some interest in an instrument for a similar purpose, which we have prepared by adding a small accessory device to a standard commercial vibrating-reed electrometer.

The complete instrument (Fig. 1) consists of a ROSSI tissue equivalent chamber*[2] connected to a vibrating-reed electrometer† with the accessory circuit in the small flat box shown at the side of the photograph. One of the normal features of the design of this particular electrometer is a high insulation relay which enables the input terminals in the head unit to be shorted by the operation of a switch in the main unit. The voltage across the input terminals is read on an instrument calibrated with full scale ranges of 10, 30, 100, 300 and 1000 mV. A recorder output giving 1 mA into 1.5 kΩ is also provided.

For use as an integrating instrument one of the high value resistors normally fitted in the head unit is removed so that the current from the ionization chamber charges the input capacity due to the chamber itself and its connexions (28 pF). The accessory circuit (Fig. 2) is a transistorized Schmitt trigger which operates when the voltage developed across the input capacity corresponds to the full scale value on any range of the main unit. When the trigger circuit operates it advances a mechanical register one unit and also operates the relay, shorting the ionization chamber. To allow for ringing and soakage effects the shorting time is made relatively long (2 sec). The sensitivity of the apparatus as shown is such that one count on the mechanical register corresponds to 0.1 mrad on the least sensitive (1000 mV) range or 1 count/μrad on the most sensitive (10 mV) range. Lower sensitivities are easily attained by connecting a low leakage capacitor in parallel with the ionization chamber.

Referring to Fig. 2, the recorder output of the electrometer is fed to the Schmitt trigger circuit (Transistors II and III) through an emitter follower (Transistor I). The triggering point may be preset on the 250-Ω rheostat, and is normally adjusted to 1.5 V corresponding to full scale deflexion of the electrometer. With switch S2 in position 1, the Schmitt circuit drives the register and closes the relay connected to the short circuiting device for 2 sec. The circuit has another possibility, that of operating as an alarm when the total dose has reached a preset value. In this application the 250-Ω rheostat is set so that the input triggers at the required dose value, then with switch S2 in position 2 the relay contacts operate an alarm, and remain closed until the circuit is reset by shorting the ionization chamber by means of a switch on the main unit of the electrometer.

The CERN 600-MeV synchro-cyclotron has a repetition rate of 55 pulses/sec so that individual radiation pulses are not resolved. Under these conditions the effect of the 2-sec dead time after each count merely shortens the apparent time of the exposure. However, the repetition rate of the 25-BeV proton synchrotron is normally 1 pulse in 3 sec, so there is an ambiguity whether a radiation pulse will occur or not during the dead time. In a long exposure a correction can be applied from elementary statistical considerations, but for most survey purposes it is more useful to read the accumulation of dose from a pen recorder chart.

The electrometer may also be used for dose measurements without modification if it is employed as a dose-rate meter with electro-mechanical integration of its output[3]. For this purpose the ionization chamber current is fed through one of the high value resistors provided in the head unit, and the recorder output is fed into an integrating motor through an amplifier used to buffer the back e.m.f. For stability and economy a simple magnetic amplifier is employed (Fig. 3). Although this method permits simultaneous readings of dose and dose rate, and does not involve periods of dead time, the time constant of the input circuit is necessarily rather long. A dose rate of

* Landis and Gyr Ltd., Zug, Switzerland.

† Electronic Instruments Ltd., Richmond, Surrey, England.

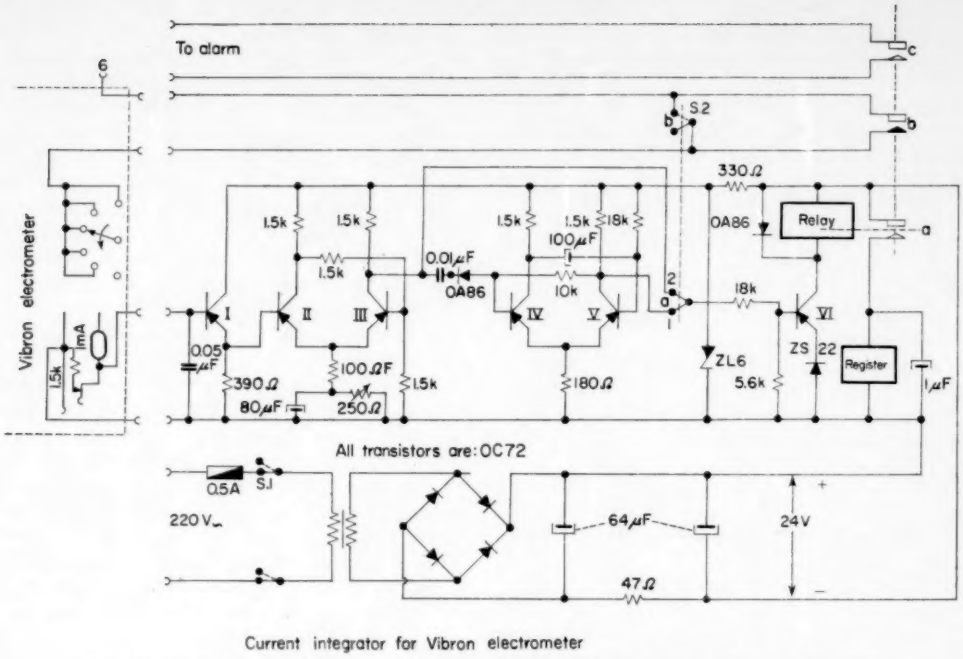

FIG. 2. Transistorized Schmitt trigger circuit for short circuiting the ionization chamber and driving a mechanical register.

0.2 mrad/hr in a Rossi chamber produces 10 mV across a 10^{12}-Ω resistor, giving full output of the electrometer on its most sensitive range. The buffer amplifier may be so adjusted that the maximum electrometer output drives an integrating motor at full speed, usually 2000 rev/min (Electromethods* Series 910). With the gear ratios found practicable between the motor and the digital drums of a revolution counter, 20 revolutions of the motor armature can be recorded as a change of one digit. Under these conditions the sensitivity is such that 1 μrad corresponds to a change of 30 digits. Alternatively, integrating motors with resettable pointers geared together like clock hands may be used (Electromethods* Series 915). The pointers are geared together with a 100:1 ratio and the instrument may be chosen to give its full count of 10,000 in 10 min, 1 hr, 24 hr or 7 days. With a 24 hr-scale, and the electrometer range set to read 10 mV, the device will read up to 4.8 mrads with a resolution of 0.5 μrad per division. Ignoring chamber saturation these sensitivities may normally be reduced by as much as 5 decades, 2 on electrometer range, and 3 on input resistance.

* Electromethods Ltd., Stevenage, Herts., England.

Acknowledgments—The Schmitt trigger unit was developed and constructed by S. O. LARSON and P. VAREKAMP in the Health Physics Group.

CERN European Organisation for Nuclear Research
Geneva, Switzerland
B. M. WHEATLEY*

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* Present address: Central Electricity Generating Board, Berkeley Nuclear Laboratories, Berkeley, Glos., England.

Simplified Autoradiography Technique for α-Emitters

(*Received* 2 *September* 1960)

Introduction

THE final step in the determination of very low concentrations of plutonium-239 in excreta and tissues is usually the autoradiography of a planchet on which

FIG. 1. The complete instruments, consisting of a TE chamber mounted on a vibrating-reed head unit, a main unit with recorder, and the accessory device described in the text.

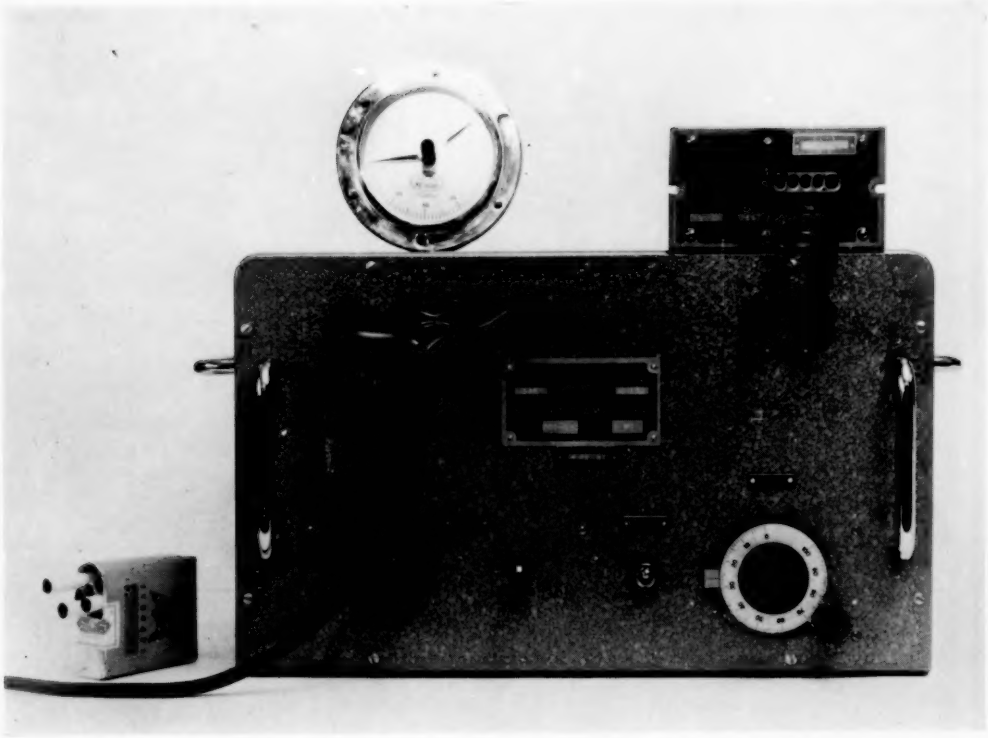

FIG. 3. Two types of integrating motor and a magnetic amplifier used as a buffer between them and the recorder socket of the electrometer.

the plutonium has been electrodeposited. A new, simplified technique for performing such autoradiographic analysis is described below. The unique features of this procedure are:

(1) Radioactive nickel-63 is used to define the area on which Pu^{239} has been electrodeposited.

(2) Cellophane tape is used to fasten the planchets (on which the Pu^{239} was electrodeposited) and a coded identification card to a Kodak nuclear track slide, type NTA, 25 μ, T-coat.

(3) The NTA slide is exposed to visible light to define the location of the planchets after development and to identify permanently the slide with a binary code.

Generally used technique

A description of the usual technique for determining Pu^{239} in urine samples may be found in Refs. 1 and 2 in the bibliography. This method of autoradiography described in the references is to use a camera for holding the planchets against the NTA emulsion during exposure. The NTA slide must be precisely located in the camera and after it is processed, it must be located precisely on the microscope or microprojector stage. Usually only part of the electrodeposited area is counted. The microscope used for reading the slide must have a calibrated stage so the correct areas may be read.

Simplified technique

A sample of excreta or tissue is processed by wet ashing with nitric acid, ion exchange separation of Pu^{239}, and finally the Pu^{239} is put into an electrodeposition cell as an ammonium nitrate–nitric acid solution at a pH of 1.2 to which has been added approximately 1100 dis/min of Ni^{63}. (Approximately 1100 dis/min of Ni^{63} per 7.9 mm^2 of plated area is used for an autoradiographic exposure time of 10,000 min.) Ni^{63} is electrodeposited along with the plutonium.

After electrodeposition, the planchet is fired in a reducing flame to fix the plutonium and nickel. Firing is done in a hood since some Ni^{63} is lost in this step. Data in Table 1 indicate that the electrodeposited Pu^{239} is not diminished in the firing step nor is it likely to be diminished by subsequent handling of the planchet.

The planchets are placed with plated side down on a clean glass slide called the storage slide (the same size as the NTA slide, 1 in. × 3 in.) together with a card punched with a binary identification number. Fig. 1(a) shows the arrangement of card and planchets on the storage slide. Cellophane tape is pressed firmly over the whole slide. The planchets are, therefore, preserved in a form easily handled or stored if exposure is not made immediately.

Table 1. *Four platinum planchets were prepared by electrodeposition of* 80 dis/min *of* Pu^{239} *on each planchet. The following counting data were obtained and are shown for* 90 *per cent confidence level*

A. *Planchets counted* before *being fired in a reducing flame*

Planchet	Counts	Min	Counts/min
1	1000	36.1	27.7 ± 1.4
2	1200	46.7	25.7 ± 1.2
3	1400	53.0	26.4 ± 1.2
4	1520	58.8	25.8 ± 1.1
			Average 26.4 ± 1.2

B. *Planchets counted* after *being fired in a reducing flame*

Planchet	Counts	Min	Counts/min
1	1830	67.5	27.1 ± 1.0
2	800	28.8	27.8 ± 1.6
3	1160	44.8	25.9 ± 1.3
4	270	10.1	26.1 ± 2.7
			Average 26.9 ± 1.6

C. *Planchets counted after being wiped rather vigorously with cleansing tissue*

Planchet	Counts	Min	Counts/min
1	1880	69.0	27.2 ± 1.0
2	1850	72.7	25.4 ± 1.0
3	1600	59.2	27.0 ± 1.1
4	1310	50.9	25.7 ± 1.2
			Average 26.3 ± 1.1

For exposure in a darkroom the tape is lifted from the storage slide and placed directly on the emulsion of an NTA slide. (Care must be taken not to press firmly on the planchets themselves because sensitization will tend to obscure the α-tracks.) The planchets and the identification card are then in intimate contact with the emulsion. [See Fig. 1(c) for a cross-section of the arrangement of planchets and slide.]

Very near the end of the 10,000-min exposure, which has been made in a light-tight box stored in a refrigerator, the slide is placed on a black, nonreflecting surface with the emulsion side up and flashed with white light [See Fig. 1(c) for relative

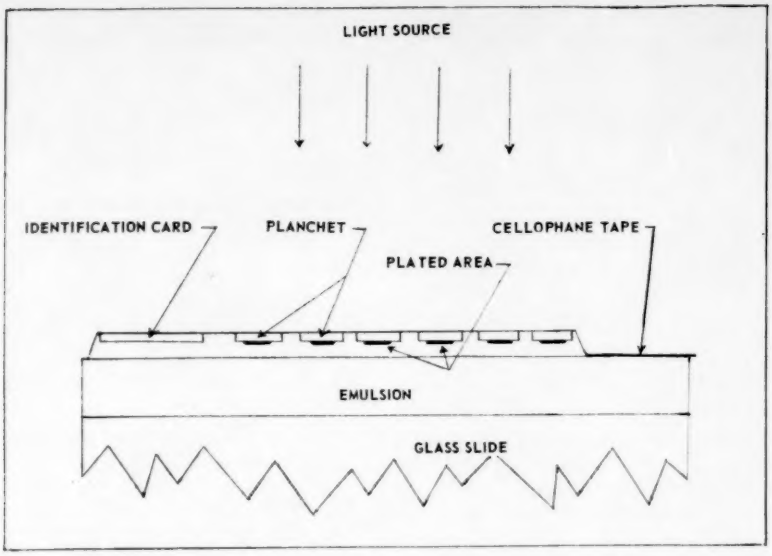

FIG. 1(c). Cross-section of slide before processing.

position of light source and slide.] Too much light is to be avoided, since reflection from the backing material can occur. Any reflected light will cause an increase in fog density and attendant difficulty in determining the plated area and reading tracks. Only enough light to define the planchet area is required.

After flashing with light and at the end of the 10,000-min exposure, the tape carrying the card and planchets is peeled off the NTA slide and fastened to a data card for permanent storage. The emulsion adheres so strongly to the surface of the glass slide that it does not come off with the tape. The exposed NTA slide is processed in the usual manner.

After development the entire slide is black except the areas under the planchets and the solid part of the identification card. [See Fig. 1(b) for a picture of a developed slide.] The exact position of each planchet and the identification code are easily discernible and become a permanent part of the slide.

Since the area around each planchet position is black, reading positions can be readily located. [See Fig. 1(d) for an enlarged picture of the identification area and one planchet area.] The plated area of the planchet is defined by the increased fog density produced by the Ni^{63} β-particles and is located with a microscope or micro-projector [See Fig. 1(e) for a picture of part of the plated area of one planchet. The arrow indicates the edge of the plated area.] Alpha-tracks in the entire plated area and only those in the plated area are counted; consequently, background tracks are reduced to a minimum without reducing the number of plutonium α-tracks that are counted. For example, the background count at NTS is normally eight tracks per electrodeposited area following a 10,000-min exposure whereas the track count for 0.06 dis/min of Pu^{239} exposed for the same length of time is 280 tracks.

This simplified autoradiography technique can be applied to the determination of extremely low levels of any α-emitter that can be electroplated along with Ni^{63}. Although it has been evaluated for only Pu^{239}, there should be no reason why it will not work for uranium and all of the transuranium elements.

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Reynolds Electrical & Engineering Co., Inc.
Las Vegas, Nevada

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† Present address: R. S. Landauer, Jr. & Co., Culver City, California.

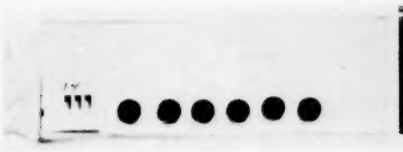

FIG. 1(a). Slide before processing.

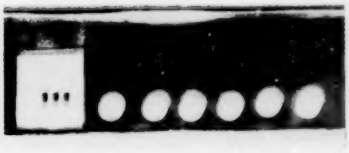

FIG. 1(b). Slide after processing.

FIG. 1(d). Enlargement of part of slide.

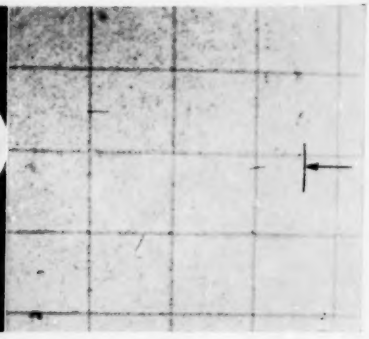

FIG. 1(e). Enlargement of part of slide.

The Deposition of Radioiodine in the Thyroids of Rats following Inhalation of the Vapour

(*Received* 29 *September* 1960)

Introduction

THE maximum permissible concentration in air for I^{131} is calculated from the behaviour of the ingested nuclide. The direct determination of the fate of inhaled iodine has not received much attention. WILLARD and BAIR[1] have reported that 2.5 per cent of the inhaled vapour and 1.6 per cent of iodine in AgI^{131} particles is deposited in the thyroid of rats. Their results are obtained from a single experiment involving several rats and no measure of the variability of individual results about the mean is given. These results differ from the retention used by the ICRP by a factor of 10 and it seemed desirable to obtain further experimental data.

Experimental

Adult males of a laboratory strain of hooded rats were used unless otherwise stated. Up to the time of exposure they were fed on "Laboratory Chow".* The normal weight of the adult males was 275–350 g and the average daily intake of food was 15–20 g. The chemical composition of the prepared food has been published in the trade literature of the supplier. The iodine content is given as 1 μg/g. Thus the normal daily intake of iodine was 15–20 μg. This is well in excess of the daily minimal uptake given as 2 μg.[3] After entering the exposure chamber, and until death, the animals had access only to water.

The exposure chamber was a wooden box 0.9 × 1.2 × 1.8 m, painted internally with Tygon. The rats were placed individually in cylindrical wire cages 5 cm in diameter and 30 cm in length, some hours before exposure to ensure that they became calm after handling. Only a few of the animals showed signs of excitability which might have affected their ventilation rate. Thirty minutes after release of the iodine, the cages were inserted into the exposure chamber through holes drilled in its sides. No attempt was made to prevent iodine vapour from reaching the body but the small diameter of the cages prevented the animals from licking their fur. The rats were kept in the wire cages after the completion of the exposure period until they were killed.

Iodine vapour was generated inside the chamber by heating carrier-free NaI^{131} in a small crucible wound with resistance wire. A small fan on the floor of the chamber produced uniform mixing. Preliminary work had shown that both particulate and gaseous iodine were initially present in the chamber, but after 30 min most of the particulate iodine had been removed by impaction on the walls and settling. The gaseous iodine concentration showed only a slow linear decrease over periods of many hours. The air in the chamber was sampled regularly during the exposure. The iodine concentration in the atmosphere could be estimated accurately, since the change during the experiment was small and linear. In each case the sampling volume was 8 l. which, being small compared with the volume of the chamber (approximately 2 m^3), did not significantly affect the conditions in the chamber. The large volume of the chamber and the small number of rats in each experiment ensured that the concentration of CO_2 would not significantly affect the ventilation rate.

The air in the chamber was sampled by aspiration through a millipore filter (type AA) and a carbon absorber similar to that described by SILL and FLYGARE[2]. The activity found on the millipore filter was assumed to be particulate in origin (other work in this laboratory has shown that the type AA millipore filter absorbs about 2 per cent of the iodine vapour in the air stream) while that on the carbon absorbers was considered to be a measure of the iodine vapour present. Samples were drawn from the chamber through tubes extending into the box to the neighbourhood of the heads of the rats. The filters were as close to the sampling head as possible to minimize the loss of iodine by adsorption on the tubing.

Results

The distribution of I^{131} in the animals as a function of time after exposure is shown in Table 1. The inhaled dose was calculated from the known concentration of I^{131} in the air, the duration of exposure and a respiration rate of 100 ml/min.[4] The results agree qualitatively with those of WILLARD and BAIR[1], but the maximum amount of iodine deposited in the thyroid was higher than that reported by these workers by a factor of 10. The organs listed in the table account for about 50 per cent of the estimated inhaled dose. Analysis of the urine excreted over a period of 24 hr after exposure accounts for over 50 per cent of the estimated inhaled dose. Hence, animals in a state of iodine balance rapidly excrete a large fraction of the inhaled dose.

The combined results of several independent experiments on adult rats are shown in Table 2. The range of variability is normal for this type of experiment.

* Manufactured by the Ralston Purina Co., St. Louis 2, Missouri.

Table 1. *Distribution of inhaled radioiodine in body of rat*

Organ	% of inhaled dose in organ as function of time after exposure			
	0 hr	24 hr	44 hr	50 hr
Thyroid	7.2	23.7	26.9	24.1
Lung	1.8	0.3	0.2	0.2
Liver	5.3	0.5	0.7	0.3
Kidney	2.0	0.2	0.3	—
GI tract	35.0	4.2	2.6	1.6

Table 2. I^{131} *in thyroids of adult* (3-month) *male rats as a function of time after exposure*

Experiment no.	No. of rats	Time after exposure (hr)	Concentration of I^{131} in air ($\mu\mu$c/ml)	I^{131} in thyroid (% of inhaled dose)	Standard deviation
1	3	0	1.64	6.5	1.04
2	1	0	0.31	7.2	—
3	1	0	0.87	7.2	—
2	1	24	0.31	44.1	—
3	2	24	0.87	32.6	—
6	5	24	1.15	36.4	5.9
10	5	24	2.09	22.8	3.3
3	3	48	0.87	27.8	3.2
4	5	48	0.35	38.7	11.4
5	5	48	0.72	22.8	3.6

Table 3. *Deposition of* I^{131} *activity in the thyroids of rats as a function of age*

Age (weeks)	Mean thyroid weight (mg)	Standard deviation	Concentration of I^{131} in air ($\mu\mu$c/ml)	Mean deposition for unit thyroid weight and unit dose after 24 hr (cm^3/mg-min)	Standard deviation
3	3.2	0.3	2.2	0.536	0.062
4	6.1	0.7	1.2	1.025	0.107
9	17.2	1.3	1.65	1.800	0.068
13	17.9	5.2	2.09	1.240	0.030
13	20.2	3.5	various	1.850	0.190

The variability is due mainly to experimental error in the determination of the concentration of I^{131} in air, the difference between experimental conditions from day to day in such factors as temperature and humidity, and variations in the rats themselves from one batch to another. The results, however, show that the present recommendations of the International Committee on Radiological Protection (1959), which assume 23 per cent retention of the inhaled dose by the thyroid, are realistic.

The effect of age was studied using rats 3, 4 and 9 weeks old. To allow for the differences in minute volume and thyroid weight with age, the results of these experiments are expressed as the activity deposited per milligram of thyroid for unit dosage. These results are given in Table 3, where the results for the adult rats are also included as a basis for comparison. When the figures in column 5 are multiplied by 100 and divided by the minute volume, one obtains the fraction of the air dose deposited in the thyroid per unit weight of organ. It is apparent from Table 3 that young animals absorb considerably less per milligram of organ than do adults, the difference being about a factor of 10 if one assumes a minute volume of about 30 ml for the 3-week-old animals.

Conclusions

In calculating the $(MPC)_a$ for iodine-131, the ICRP assume that 75 per cent of the inhaled iodine is absorbed by the body and 30 per cent of the absorbed dose is deposited in the thyroid. The overall efficiency is therefore 23 per cent. The results of this study are in reasonable agreement with this, particularly in view of the uncertainties in the physiological factors that control the deposition of iodine in the thyroid.

Acknowledgement—Mr. P. C. Jay provided valuable assistance by performing most of the experiments described.

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Note on the ICRP and NCRP Recommended $(MPC)_a$ for Insoluble Ra^{226} Compounds

(*Received* 15 *August* 1960)

In the current publications of NCRP[1] and ICRP[2], the $(MPC)_a$ of 2×10^{-7} $\mu c/cm^3$ for insoluble compounds of Ra^{226} for a 40-hr work week is based on the g.i. tract as the critical organ. The lung also was considered as the critical organ, but the calculation was deferred in the hope of obtaining better data on the retention of Rn^{222} in the lung. Such data were not found, and a value based on lung was inadvertently omitted from the final manuscript.[3] For other isotopes of Ra, i.e. Ra^{223}, Ra^{224} and Ra^{228}, a $(MPC)_a$ value for insoluble compounds is listed for both the lung and the g.i. tract. It is the purpose of this note to make use of data recently published by ICRP[2] and other sources of information to compute the $(MPC)_a$ value based on lung as the organ of reference for the case of insoluble Ra^{226} compounds and to show that the lung should be taken as the critical organ.

To calculate this $(MPC)_a$ we combine equations (4) and (7) appearing in the Committee II report of ICRP[2] and obtain

$$(MPC)_a = \frac{8.4 \times 10^{-11}\ m}{\varepsilon T f_a (1 - e^{-0.693t/T})} \qquad (1)$$

Inserting the values $m = 1000$ g, $T = 120$ days, $f_a = 0.125$ and $t = 50$ years (18,250 days) into (1), we find

$$(MPC)_a = \frac{5.6 \times 10^{-9}}{\varepsilon}\ \mu c/cm^3 \qquad (2)$$

The effective energy per disintegration, ε, is calculated from equation (35) of the Committee II report of ICRP, and the values for the parent and daughter isotopes in a given organ of reference are tabulated and appear in Table 5(a) of this report. No values are shown in Table 5(a) for lung as the organ of reference for the Ra^{226} chain. These values are presented in Table 1.

In calculating the total chain effective energy, it was assumed that 30 per cent of the Ra^{226} daughters

Table 1

Radionuclide	Half-life (T_r in days)	Organ of reference	ΣE(RBE)n (E in MeV)	F	ΣEF(RBE)n (E in MeV)
$_{88}Ra^{226}$	5.9×10^5	Lung (insol.)	49.27	1	49.27
Rn^{222} through Tl^{210}	3.83	—	200.42	0.30	60.13
Pb^{210}	7.1×10^3	—	0.027	0.0050	1.4×10^{-4}
$Bi^{210} + Tl^{206}$	5.0	—	0.40	0.0048	0.0019
Po^{210}	138.4	—	55.04	0.0022	0.12
			Lung (insol.)		Total for chain 110

is retained in lung tissue. This assumption will be discussed later.

From Table 1 we find

$$\varepsilon = 110 \text{ MeV} \tag{3}$$

and from (2) and (3) we calculate

$$(\text{MPC})_a = \frac{5.6 \times 10^{-9}}{110} = 5 \times 10^{-11} \ \mu\text{c/cm}^3 \tag{4}$$

This value is very much less than the 2×10^{-7} μc/cm^3 value based on the g.i. tract as the organ of reference. It, therefore, requires that the lung be taken as the critical organ for exposure to insoluble compounds of Ra^{226}. However, this value depends on the assumed retention of Ra daughter products in lung tissue.

Thirty per cent retention of Ra daughter products is known to apply to the case of Ra stored in skeleton. Some investigators are of the opinion that some value less than 30 per cent would apply to the case of lung-stored Ra.[4,5] There are no experimental measurements available from which a retention value for the Rn^{222} daughter can be determined and agreed upon. However, for purposes of comparison, we can calculate the $(\text{MPC})_a$ assuming 0 per cent retention of Ra daughter, i.e. all of the Rn formed from Ra stored in lung tissue is exhaled. In this case we take for the value of ε

$$\varepsilon = 49 \text{ MeV} \tag{5}$$

the effective energy of the Ra^{226} parent appearing in Table 1. Then from (2) and (5) we calculate

$$(\text{MPC})_a = \frac{5.6 \times 10^{-9}}{49} = 1 \times 10^{-10} \ \mu\text{c/cm}^3 \tag{6}$$

This is only a factor of 2 greater than the $(\text{MPC})_a$ calculated in (4).

Regardless of whether 0 or 30 per cent of the daughters are retained in lung tissue for lung-stored Ra, these calculations indicate that the lung should be considered as the critical organ. For the present, it is suggested that 30 per cent retention of Ra daughters be adopted for lung-stored Ra, pending better data. Consequently, it is proposed that $(\text{MPC})_a$ for insoluble Ra^{226} compounds be set at 5×10^{-11} μc/cm^3 rather than 2×10^{-7} μc/cm^3 for a 40-hr-week exposure and 2×10^{-11} μc/cm^3 rather than 6×10^{-8} μc/cm^3 for a 168-hr-week exposure.

Acknowledgment—Dr. F. Western of the USAEC called this problem to the attention of Dr. W. S. Snyder who, in turn, suggested this note be written by the authors.

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* Operated by Union Carbide Corporation for the U.S. Atomic Energy Commission.

Some Remarks on the Maximum Permissible Concentration of Unidentified Radionuclides in Water and Related Disposal Formulae

(*Received* 6 *June* 1960)

In Table 3 of the Report of Committee II of the ICRP[1] values are given for the maximum permissible concentration of unidentified radionuclides in water, $MPCU_w$. These values are based on grouping the nuclides from the new recommended values of the maximum permissible concentration in water for 168 hr/week occupational exposure, given in Table 1 of the same report. For interim application in the neighbourhood of an atomic energy plant one-tenth of these values are used.

Consider four groups A, B, C and D of nuclides, composed as follows:

Group A, Ra^{226} and Ra^{228}.

Group B, Sr^{90}, I^{129} and Pb^{210}.

Group C, Po^{210}, Ra^{223}, Pa^{231} and Th-nat.

Group D, I^{126}, I^{131}, At^{211}, Ra^{224}, Ac^{227}, Th^{230} and Th^{232}.

The recommended provisional values of $MPCU_w$ are:

3×10^{-5} $\mu c/cm^3$ on the condition that the nuclides of the groups A, B, C and D are not present (i.e. the concentration is very much $< MPC_w$);

2×10^{-5} $\mu c/cm^3$ if A, B and C are not present;

7×10^{-6} $\mu c/cm^3$ if A and B are not present;

10^{-6} $\mu c/cm^3$ if A is not present;

10^{-7} $\mu c/cm^3$ with no restricting condition.

For the derivation of these values we consider only the lowest MPC_w value given for any nuclide, soluble or not. Table 1 gives the nuclides having MPC_w values $\leqslant 10^{-4}$ $\mu c/cm^3$.

Considering one of the vertical lines in Table 1, the next higher concentration is allowed, provided the radionuclides left of the line are not present. For example the line between I^{129} and Po^{210} indicates that 7×10^{-6} $\mu c/cm^3$ is allowed provided that Ra^{226}, Ra^{228}, Sr^{90}, Pb^{210} and I^{129} are not present. For the derivation of the $MPCU_w$ values in the ICRP report only a few vertical lines are used, viz. the dashed ones.

The provisional $MPCU_w$ values given in Table 3 of NBS Handbook 69[2] are somewhat different. During the preparation of this table the MPC_w-values of the I-isotopes were taken a factor 1.5 too high. The writer was informed that the errors were discovered before the issue of the NBS Handbook, but too late to get the change into the published version. The issue of an errata sheet is planned.

A somewhat simplified system of $MPCU_w$ values can be derived from Table 1 as follows. We introduce two new groups E and F of nuclides to replace the groups C and D, viz.:

Group E, Po^{210} and Ra^{223}.

Group F, all α-emitters and radioactive iodine nuclides not included in the groups A, B and E.

This leads to the following $MPCU_w$ values:

10^{-4} $\mu c/cm^3$ on condition that no nuclides of the groups A, B, E and F are present;

10^{-5} $\mu c/cm^3$ if A, B and E are not present;

7×10^{-6} $\mu c/cm^3$ if A and B are not present;

10^{-6} $\mu c/cm^3$ if A is not present;

10^{-7} $\mu c/cm^3$ with no restricting condition.

The following remarks have to be made concerning this suggestion:

(1) No MPC_w values would have been surpassed if Pa^{231} would have had a MPC_w of 10^{-5} $\mu c/cm^3$ instead of the actual value 9×10^{-6} $\mu c/cm^3$, and Ca^{45} and Cs^{134} a value of 10^{-4} $\mu c/cm^3$ instead of 9×10^{-5} $\mu c/cm^3$. However, the inaccuracies of the recommended MPC values are certainly much more than 10 per cent, so that on this basis an underestimation of the danger by about 10 per cent is quite acceptable.

(2) The $MPCU_w$ values suggested here are *less* restricting than those recommended by the ICRP for Pa^{231}, Th-nat (10^{-5} $\mu c/cm^3$ instead of 7×10^{-6} $\mu c/cm^3$) and for all non-α-emitters except Sr^{90}, I^{126}, I^{129} and I^{131} (10^{-4} $\mu c/cm^3$ instead of 3×10^{-5} $\mu c/cm^3$); *more* restricting for I^{126} and I^{131} (10^{-5} $\mu c/cm^3$ instead of 2×10^{-5} $\mu c/cm^3$) and all α-emitters except Pb^{210}, Po^{210}, Ra^{223}, Ra^{226}, Ra^{228}, Pa^{231} and Th-nat (10^{-5} $\mu c/cm^3$ instead of 2×10^{-5} $\mu c/cm^3$, resp. 3×10^{-5} $\mu c/cm^3$).

The formula used at Harwell for disposal in the river[3] is based on grouping MPC_w values recommended in former days. This formula is:

$$2500(\text{Ra}) + 420(\alpha\text{-emitters}) + 50(\text{Sr}^{89} \text{ and } \text{Sr}^{90}) + (\beta\text{-emitters}) \leqslant 10^{-4}k$$

in which by the nuclide in brackets is meant the activity in curies disposed per month; 10^{-4} c/m^3 is an approximate lower limit of the MPC_w values of β-γ-non-α-emitters and k m^3/month is a disposal factor that includes the water flow through the river and a safety factor. In Harwell $k = 2.6 \times 10^6$ V/S, in which V is the water flow through the river (m^3/sec) and S the safety factor, so that, after completely mixing, the concentration in the river $\leqslant MPC_w/S$. For Harwell S has been taken 100. Comparison of the disposal allowances of other institutes with the Harwell formula gives the following values of S (within the brackets the country and the

Table 1. *Nuclides ordered following their* 168-hr MPC_w *values*

α-Emitters are underlined, solid if the nuclide itself is an α-emitter, dashed if there is some important α-emitter only in the chain[1] [p.8(d) and Table 1]. Groups are indicated. The ICRP grouping of $MPCU_w$ values are based on the dashed vertical lines, the grouping suggested in this paper on the solid vertical lines.

μc/cm³	10^{-7}(A)	3×10^{-7}(A)	10^{-6}(B)	4×10^{-6}(B)	7×10^{-6}(C; E)	9×10^{-6}(C; F)
nuclide	Ra^{226}	Ra^{228}	Sr^{90}; Pb^{210}	I^{129}	Po^{210}; Ra^{223}	Pa^{231}

μc/cm³	10^{-5}(C; F)	2×10^{-5}(D; F)	3×10^{-5}(F)	
nuclide	Th-nat.	I^{126}; I^{131}; At^{211}; Ra^{224}; Ac^{227}; Th^{230}; Th^{232}	Np^{237}	

μc/cm³	4×10^{-5}(F)	5×10^{-5}(F)
nuclide	Am^{241}; Am^{243}; Cm^{245}; Cm^{246}; Cf^{249}	U^{230}; Pu^{238}; Pu^{239}; Pu^{240}; Pu^{242}; Cm^{243}

μc/cm³	7×10^{-5}(F)	9×10^{-5}	10^{-4}	
nuclide	I^{133}; Th^{228}; Cm^{244}; Cf^{252}	Ca^{45}; Cs^{134};	Sr^{89}; Ru^{106}; Ce^{144}; Cf^{250}	

river of disposal): Argonne Nat. Lab. (U.S.A.; Des Plaines River) $S = 40$; Mol (Belgium; Nethe) $S = 100$; Marcoule (France; Rhone) $S = 1400$; Knoll's Atomic Power Lab. (U.S.A.; Mohawk River) $S = 2000$; KEMA (Netherlands; Rhine) $S = 65{,}000$.

These values of S are calculated from water flow and maximum amount of radioactivity allowed to dispose as mentioned in the literature.[4–8] These great differences in S mainly reflect more our lack of knowledge than the differences in rivers, surroundings and water utilization. Though the use of a formula of this kind is questionable,* it is possibly the best we have for the moment. Anyhow it is widely used. Though the urgently needed knowledge to improve the factor k is not yet available, it is advisable, where possible, not to neglect the improvement of the other part of the formula.

Using the latest MPC_w values and the groupings of the radionuclides as mentioned above, one can find the following revised Harwell formulae:

(1) from the ICRP grouping

$$300(\text{A}) + 30(\text{B}) + 4.3(\text{C}) + 1.5(\text{D}) + (\text{rest}) \leqslant 3 \times 10^{-5}k$$

(2) from the grouping suggested in this paper

$$1000(\text{A}) + 100(\text{B}) + 14(\text{E}) + 10(\text{F}) + (\text{rest}) \leqslant 10^{-4}k$$

or in full:

$$1000(Ra^{226} + Ra^{228}) + 100(Sr^{90} + I^{129} + Pb^{210}) + 14(Po^{210} + Ra^{223}) + 10(\text{other } \alpha\text{-emitters} + \text{other iodine isotopes}) + (\text{rest}) \leqslant 10^{-4}k$$

This formula, though in a somewhat different edition, was already communicated during the discussions on R. H. Burns' paper at the IAEA Conference on Disposal of Radioactive Wastes, Monaco, November 1959.

A series of other formulae can be derived by different ways of grouping the MPC_w-values. The choice will depend on the local situation, e.g. the procedure of control and the radionuclides that are important. The formula suggested in this article will be preferred where pure β–γ-emitters are the most important radionuclides and has the advantage of a more simple determination of the applicable level as all α-emitters are included in group F.

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* Instead of basing the formula on grouping of MPC_w values, in future it may be better based on grouping of maximum permissible total relative intake concentrations RIC, defined as

$$\text{RIC} = (I_w / \sum_i I_i f_i) \times \text{MPC}_w$$

$\sum_i$ denotes summation over all i biological chains along which the considered disposed radionuclide can enter the human body. f_i is the concentration factor along the ith chain. I_w and I_i are the average daily intakes of water and the end product of the ith chain, respectively. The summation includes the direct water intake, for which $I_i = I_w$ and $f_i = 1$.

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A Method of Bulk Liquid Evaporation for Fallout Analysis

(*Received* 23 *May* 1960)

DURING the early stages of monitoring rainwater for fallout at this laboratory, it was realized that the system using successive evaporations and transfers into smaller containers could lead to variable losses of material. A trial water-dropper was developed which allowed the water sample to drop slowly on to a small continuously heated container which was subsequently counted. This method was not pursued, however, because the small orifice through which the water emerged frequently became blocked by foreign material suspended in the water.

The author concurrently developed the plastic-sheet method. This consists of lining a metal evaporating dish with a thin Polythene film, 0.0015 in. thick, introducing the water sample and evaporating to dryness. The plastic sheet is then folded in a small metal tin (0.4 × 1.4 in. diameter) and carefully heated from below and cooled from above. The plastic melts, ignites and burns slowly away, leaving very little ash, which is counted in the same container.

It has been found that a constant level waterbath is a suitable method of heating the evaporating dish. It helps prevent the plastic from overheating and "catching" to the dish and requires very little attention. The plastic sheet is fixed in position by either clips or by first heating the lip of the evaporating dish, and pressing the sheet into position especially round the lip, where the plastic softens and just slightly attaches itself. When the evaporation is completed, the plastic sheet is removed and trimmed to size around the imprint left in the plastic by the metal lip.

Two evaporating dishes over a single waterbath holding a total of $\frac{3}{4}$ gal can be conveniently evaporated in $6\frac{1}{2}$ hr; further liquid may be introduced into the same piece of plastic as required. This evaporator has been routinely in use for three years and a second model with a single evaporating dish has been used routinely for two months. This single dish holds $\frac{1}{2}$ gal and is conveniently evaporated in $4\frac{1}{2}$ hr. More recently the same technique has been used successfully for fresh-milk samples.

Tests with spiked solutions have shown that the over-all effect of material retained on the walls of the tin has a negligible effect on the count rate and that the recovery is better than 95 per cent.

The dimensions of the equipment are not critical; those mentioned are currently in use as they were fairly readily available. Experience has shown that this is a simple, efficient, reliable system which is capable of more general application for radioactive assay.

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LETTER TO THE EDITOR

(*Received* 18 *July* 1960)

THE paper "Electrodeposition of Uranium from Urine"[1] brought to mind an observation which was made during the testing of this technique at Atomics International. Urine specimens, spiked with uranyl nitrate hexahydrate, were prepared and electrodeposited according to the procedure reported by FRED WILLIAMS in NLCO-595. The electrodeposition unit used was apparently the same as used by ROYSTER[1].

A series of specimens spiked to four different

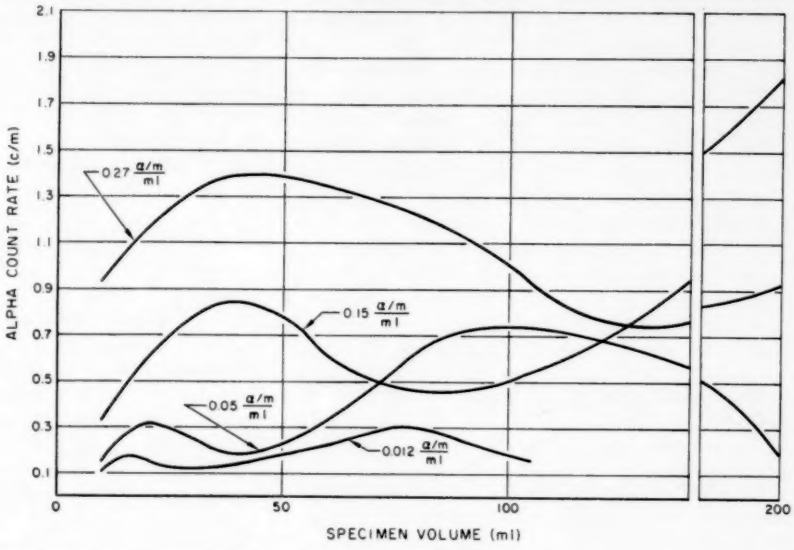

FIG. 1. Volume to yield optimum α-count rate for various uranium concentrations.

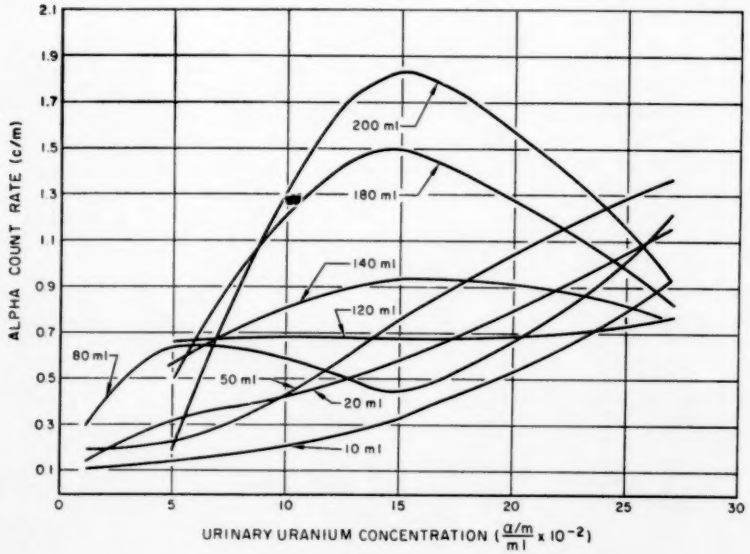

FIG. 2. Effect of increased urinary uranium concentration on α-count rate for various specimen volumes.

normal uranium concentrations was processed to determine the specimen volume which should be used in order to obtain the maximum α-count rate. As shown in Fig. 1, the count rate as a function of volume, with constant concentration, is not single valued, and the danger of a low interpretation does exist if the specimen volume is too large. Fig. 2, a re-plot of the same data, indicates that the urinary uranium concentration is uniquely related to the count rate only for small specimen volumes. However, a larger volume may be selected to increase subsequent count rates if the magnitude of the concentration is first estimated with a small specimen.

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NEWS

News Editor: J. J. FITZGERALD

Reorganization of AEC licensing and regulations

The Division of Licensing and Regulations has been reorganized by the AEC and the three new assistant directors are as follows: Mr. E. R. PRICE, Assistant Director, Administrative Branch; Mr. L. E. JOHNSON, Assistant Director, Facilities and Materials Licensing; and Mr. C. K. BECK, Assistant Director, Nuclear Facilities Safety. Mr. H. L. PRICE is the Director of the Division and Mr. R. L. KIRK is the Deputy Director.

Proposal for reactor at New York Fair canceled

Officials of the New York World's Fair for 1964 reported that on the basis of an opinion rendered by the AEC the plans to use and to exhibit a nuclear power plant at the Fair have been turned down. It has been reported that the AEC statement was to the effect that "the state of the art is not sufficiently advanced to permit placing a power plant at the World's Fair". The World's Fair officials stated that they understood this decision to mean that the AEC evaluation staff "disapproved the project on safety grounds". It was noted that in reference to this decision, the ship-mounted nuclear plant for the Fair would have been smaller than the 70 tMW reactor powering the N.S. Savannah, which is scheduled to be operated in the densely populated Camden, New Jersey area next year.

U. S. Coast Guard regulations

In October, 1960, the U.S. Coast Guard published regulations pertaining to nuclear shipping in the Federal Register, Part 57, Title 46—Shipping. The regulations are considered general in nature and they authorize the Commandant of the Coast Guard to prescribe the detailed requirements applicable to each vessel pending the development of specific safety standards. The new regulations which are presently in effect include:

(1) Annual Coast Guard Inspection of the Nuclear Vessel.

(2) Adherence to 10 CFR 20 (AEC's "Standards for Protection Against Radiation").

(3) Prevention of "the uncontrolled release of hazardous amounts of radioactive material to normally accessible spaces or to the ship's environment in the event of an accident".

(4) Provision of "standby and emergency components essential to safe operation".

(5) Assurance of compliance with applicable AEC requirements..

Studies of AEC reactor licensing; regulatory functions and organization are being conducted

A study of AEC licensing of reactors is being made by DAVID TOLL, Staff Counsel of the Joint AEC Committee, together with Consultants, DAVID CAVERS, Harvard Law School, and WILLIAM MITCHELL, former AEC general counsel. All reactor problems which have come to public hearing are being studied and the AEC staff and license applicants have been or are being interviewed. The regulatory functions and the organization of the AEC is being studied by an AEC task force under the direction of WILLIAM SLATON, General Manager for Administration. A similar study is being made by the co-directors of the University of Michigan Law School Atomic Energy Research Project, WILLIAM BERMAN and LEE HYDEMAN.

Request for public hearings on reactor siting

A proposal to have public hearings on nuclear site criteria in 1961 was sent to Representative HOLIFIELD, chairman of the joint committee on Atomic Energy, by OLIVER TOWNSEND, director of the New York State Office of Atomic Development. TOWNSEND stated the need for a "clear national policy on the safe siting of such major atomic energy installations as power generating stations, process heat reactors and nuclear port facilities".

TOWNSEND also said that "the lack of a clear siting policy is also currently complicating consideration of the Port of New York as a service port for the *N.S. Savannah*".

Problems associated with waste disposal by air (British)

At the meeting of the International Union of Aviation Insurers that was held in Sicily, Mr. R. H.

JENNENS of the British Aviation Insurance Company, stated that "British Aviation Insurance felt that it was against public interest to transport nuclear waste by air because of the hazards involved in a possible incident". He stated, however, that "there were no known instances of any such radioactive waste materials having been transported by air over Britain".

French radioactive waste disposal problem in the Mediterranean

Widespread public opposition to the French AEC radioactive waste program was encountered when an announcement was made that the French AEC intended to dispose of 6500 barrels of low-level radioactive waste at a depth of 7800 ft in the Mediterranean. The proposed site was approximately 50 miles off the shore of France, midway between the Island of Corsica and the French Riviera. The total activity involved would be approximately 400 c. The French AEC stated that the disposal involved no danger and that the procedures were in adherence to the recommendations set forth by the International Atomic Energy Agency. Opposition to the disposal included Prince RAINIER of Monaco.

The Commissariat has indicated that despite the unfavorable comments on the planned disposal of radioactivity in the Mediterranean, the disposal is a safe operation. Furthermore, the disposal will be made when, in the opinion of the High Commissioner of Atomic Energy, adequate contact has been made with persons now lacking sufficient information to allay their fears concerning this disposal. The Commissariat noted that all qualified authorities including the Minister of Public Health have given their approval for this disposal.

Ford Foundation grants received by ICRP and ICRU

Grants have been made by the Ford Foundation to the International Commission on Radiological Protection ($250,000), the International Commission on Radiological Units and Measurements ($185,000) and the International Bureau of Weights and Measures ($32,500) to assist in "defining, measuring and limiting hazards resulting from man-made radiation". The grants were given in essence to aid these organizations in the development of their functions and to assist in being independent of the policies and pressures of governments.

Department of Justice replies to Y-12 criticality victim claimants

In reference to the law suit against the AEC that has been introduced by the eight employees of Union Carbide Nuclear Company, who were involved in a criticality incident at the Y-12 Area, the U.S. Department of Justice, in part, filed the following answers to the questions of the claimants.

(1) The occurrence of the criticality incident was admitted but there was a denial that the chemicals, gas, energies, metals and substances (involved) were of an inherently dangerous character.

(2) The employees were not employees of the AEC but rather of Union Carbide Company and that the case does not come under the U.S. Court but is rather governed by the Workmen's Compensation Law of the State of Tennessee.

(3) It was also stated that "Any injuries sustained or suffered by the plaintiff arising out of or in connection with his employment by Union Carbide Nuclear Company at the Y-12 Area mentioned by the Complainant were caused in whole or in part, or were contributed to by the negligence or fault or want of care of the plaintiff and not by any negligence or fault or want of care on the part of the defendant".

In essence, the answers assert that the plaintiff in accepting employment at the Y-12 Plant, assumed the risk of such injuries.

Radiation incident claimant and family take physical examination at Mayo Clinic

An offer of an expenses-paid physical examination was made by the AEC to JACKSON MCVEY who was involved in an incident while working with Ir^{192} for the M. W. Kellogg Company in Texas. This offer has been accepted and MCVEY, his wife and two children will be examined at the Mayo Clinic in Rochester, Minnesota. The findings of the examinations will be sent to both the AEC and MCVEY. The AEC reserves the right to make the results a part of the public record. In addition, the AEC's offer of support for the physical examination was made "apart from any consideration of liability".

Harvard University conducts Air Cleaning Institute

An Air Cleaning Institute sponsored by the American Industrial Hygiene Association was conducted at the Harvard School of Public Health during the week of January 30, 1961. The course included:

(A) Basic Theories and Principles of Aerosol and Gas Collection: particle dynamics and behavior.

(B) Stack Sampling Methods and Data Evaluation: loading, particle size, efficiency.

(C) Performance of Typical Air and Gas Cleaners: efficiencies, costs, applications to special industrial hygiene and air pollution problems.

(D) Laboratory Demonstrations: illustrations of lecture material.

World Health Organization Fellowships offered

At the request of the United States Government, the World Health Organization will provide to United States citizens in 1961 a limited number of short-term fellowships for the "improvement and expansion of health services". All applications will be submitted to a World Health Organization Fellowship Selection Committee established in 1959 by Surgeon General BURNEY. Applications will be considered in public health and allied fields. Applicants must be engaged in full-time public health or educational work. In making selections, the Committee will consider the ability of the individual and the importance of the contribution which his foreign study will make on his return. Officers and employees of the United States Government are not eligible.

The fellowships will cover *per diem* and transportation, and, except in very unusual circumstances, will be limited to short-term travel purposes, i.e. two to four months. Employers of successful applicants will be expected to endorse applications and to continue salary during the fellowship. Successful applicants could probably not start their fellowships before May 1, 1961. Deadline for receipt of applications is February 1, 1961. Further information and appropriate application forms may be obtained from Dr. HOWARD M. KLINE, Secretary, World Health Organization Fellowship Selection Committee, Public Health Service, Washington 25, D.C.

Changes in positions announced

EDGAR C. BARNES has been appointed manager of radiation protection for Westinghouse Electric Corporation's atomic power group, and will serve as co-ordinator and consultant for the atomic power group in the general areas of industrial hygiene, health physics, radioactive waste disposal and nuclear safety.

Mr. ROBERT G. GALLAGHER has joined the Nuclear Science and Engineering Corporation of Pittsburgh as an Assistant Manager of the Biology and Medicine Department and will direct Health and Safety programs.

JOHN W. HEALY has been appointed Consultant Technological Hazards of the General Electric Company. In this newly established position in Engineering Services, Mr. HEALY will be available to all General Electric components for consultation regarding technological hazards.

AEC formalizes safety and licensing requirements for radiography

The AEC has issued a new regulation and an amendment to an existing regulation to establish specific safety and licensing requirements for persons using sealed sources of by-product material (radioisotopes) in radiography.

The new regulation, entitled "Radiation Safety Requirements for Radiographic Operations", will become Part 31 of Title 10, of the Code of Federal Regulations. The new amendment will become part of Part 30, Title 10, of the Code of Federal Regulations, entitled "Licensing of Byproduct Materials". The regulation becomes effective 90 days, and the amendment 30 days, after publication in the Federal Register on November 29, 1960.

Licenses to use by-product materials in radiography are currently issued under the general provisions of AEC regulations, and applicants are required to submit complete and detailed descriptions of their operations for inclusion in their licenses.

The regulation sets up rules for equipment control, including levels of radiation for radiographic exposure devices and storage containers, the locking of radiographic exposure devices, storage precautions, use of radiation survey instruments, leak testing of sealed sources, inventory and utilization logs; personnel monitoring; and precautionary procedures in radiographic operations.

The amendment to Part 30 establishes the requirements and procedures by which an applicant may obtain a specific license for radiographic work in which he proposes to use sealed sources.

AEC publishes first booklet of series on biological and medical research

The Atomic Energy Commission has published the first in a series of pamphlets describing its life sciences research program. The purpose is to acquaint scientists, students and interested members of the public with the bio-medical program, its objectives and needs. The Commission's bio-medical program is directed toward the accumulation of knowledge of the effects of nuclear radiation from any source—natural or man-made—upon living things.

The booklet, titled "Marine Sciences Research" (TID 4040), was prepared under the direction of the Commission's Division of Biology and Medicine. It summarizes work in progress at thirteen institutions through twenty-eight research contracts in the following major areas: (1) studies in and around waste-disposal areas; (2) circulation and mixing in estuaries and coastal waters; (3) open ocean research; (4) sedimentation; and (5) distribution of elements by organisms.

The second and third booklets in this series will feature the Commission's "Genetics Research" and "Cancer Research" programs. Later additions to this series will cover other portions of the Commission's bio-medical research program.

Industrial users of radioisotopes continue to increase

The Deputy Director of the Office of Atomic Development in New York State, JON D. ANDERSON announced that the number of industrial users of radioisotopes has increased 10.5 per cent (from 161 to 178) over the last nine months. During the previous 18-month period, the increase in New York State was 7.0 per cent.

Over the last nine months, USAEC data indicated increases of 9.5 per cent in California and Texas, 7.5 per cent in Illinois and 0 per cent in Pennsylvania. The averaged rate of increase for the United States during the same period (which reflects the rapid rate of growth of some of the previously atomically undeveloped areas) was 10 per cent.

Announcement of the International Congress of Biophysics

An International Congress of Biophysics will be held in Stockholm from July 31 to August 4, 1961. The purpose of the meeting is to provide a forum for international communication in the field of biophysics. Participants may include members of national societies of biophysics, medical physics, and related fields; and other scientists interested in pure and applied biophysics. The meeting will be divided between a series of symposia devoted to special topics in biophysics and to presentations of a number of contributed papers in pure and applied biophysics submitted by the participants. Further information can be obtained from Dr. Bo LINDSTRÖM at the Department of Medical Physics, Karolinska Institutet, Stockholm 60, Sweden.

National Committee on Radiation Protection and Measurements

Just recently two new organizations have been accepted by the NCRP as sponsors. These are the American Nuclear Society and the Genetics Society of America.

Another action taken recently involved the reactivation of Subcommittee 1, "Permissible Dose from External Sources". This subcommittee under the chairmanship of H. M. PARKER met on October 18, 1960 and began an active program, reflecting its concern with the development of the basic concepts and philosophy of radiation protection. The subcommittee decided to operate under a pattern found to be successful with other subcommittees, and will have a small working group assisted by an advisory group or consultants, as circumstances demand. The initial membership includes:

H. M. PARKER, Chairman	H. F. FRIEDELL
R. H. CHAMBERLAIN	H. J. MULLER
J. F. CROW	C. POWELL
H. CURTIS	H. H. ROSSI

Subcommittee 14 on "Permissible Exposure Doses under Emergency Conditions" under the chairmanship of G. V. LEROY has been very active of late, and report of the subcommittee directed particularly to the problems of civil defense officials is nearing completion.

Subcommittee M-4 on "Relative Biological Effectiveness" under the chairmanship of V. P. BOND has begun a very active program. They have outlined a method of approach for their program and begun preliminary work on a report. Arrangements have been made for liaison between this subcommittee and the joint ICRU–ICRP group concerned with similar problems.

Two new handbooks have been released during recent months and a number of others are now nearing completion. NBS Handbook 72* on "Measurement of Neutron Flux and Spectra for Physical and Biological Applications" was released on July 15, 1960. Handbook 73 on "Protection Against Radiations from Sealed Gamma Sources" was released on July 27, 1960. Handbook 75 on "Measurement of Absorbed Dose of Neutrons and Mixtures of Neutrons and γ-Rays" and Handbook 76, "Medical X-rays Protection up to Three Million Volts" (Revision of Handbook 60), are now at the printers and should be released soon. Two other handbooks are now undergoing editorial review and can soon be released. These are: "Stopping Powers for Use with Cavity Chambers" and "A Manual of Radioactivity Procedures".

New England Chapter of Health Physics Society organizes

The New England Chapter of the Health Physics Society, comprised of health physicists and representatives from allied professions concerned with control of the hazards of ionizing radiation, held a dinner meeting at the MIT Faculty Club on November 1, 1960. Dr. WALTER D. CLAUS, President-Elect of the Health Physics Society, spoke on the subject, "Professional Status for the Health Physicist". Professor J. J. FITZGERALD of Harvard University is Chairman of the New England Chapter.

AEC engineering materials available in blueprint form

The AEC announces a new method of making engineering materials (drawings, specifications, photographs, and bills of materials) available to the general public at *greatly reduced costs* compared with

* National Bureau of Standards Handbooks are for sale by the superintendent of Documents, U.S. Government Printing Office, Washington 25 D.C.

previous procedures. Contractual arrangements have been completed with

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Individuals, industrial firms, and government agencies and their contractors are advised to communicate directly with the Cooper-Trent Corporation in regard to prices of complete packages or sets of CAPE-numbered materials, or specific drawings within packages, that have been announced in the *Engineering Materials List*, TID-4100 and Supplements.

The Office of Technical Information Extension, P.O. Box 62, Oak Ridge, Tennessee, will continue to prepare and distribute, free of charge, Supplements to the *Engineering Materials List* to announce new materials that are available.

This catalog describes the contents of the sets or packages. An abstract is included to explain the uses of the equipment or facility, in addition to a detailed list of drawings and other materials that constitute the packages. Indexes for the catalog are also provided.

Inquiries regarding materials not announced in the *Engineering Materials List* should be addressed to Office of Technical Information Extension. This Office will obtain the materials from the originating organization and will inform the requester of their availability from the Cooper-Trent Corporation.

HEALTH PHYSICS SOCIETY

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